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Idaho National Engineering Laboratory Historical Dose Evaluation

Volume 1

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IDAHO NATIONAL ENGINEERING LABORATORY HISTORICAL DOSE EVALUATION

VOLUME 1

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ABSTRACT

The methodology and results are presented for an evaluation of potential radiation doses to a hypothetical individual who may have resided at an offsite location with the highest concentration of airborne radionuclides near the Idaho National Engineering Laboratory (INEL). Volume 1 contains a summary of methods and results. Volume 2 contains more detailed discussions of methods, data, results, assumptions, and citations of reports and reference material. The years of INEL operations from 1952 to 1989 were evaluated. Radiation doses to an adult, child, and infant were estimated for both operational (annual) and episodic (short–term) airborne releases from INEL facilities. Atmospheric dispersion of operational releases was modeled using annual average meteorological conditions. Dispersion of episodic releases was generally modeled using actual hourly wind speed and direction data at the time of release.

Estimated radiation doses were within the radiation protection standards applicable at that time for every year of INEL operation. In general, potential offsite doses have been small. Doses from operational and episodic releases during the late 1950s may have been as high as 9% of the applicable whole body dose standard in effect at that time and 90% of the standard for doses to individual organs of the body. In addition, the trend in recent years has been toward smaller doses as total INEL Site releases have declined. For the more restrictive standards in force from 1985 to 1989, offsite doses to a hypothetical maximally exposed individual have been about 1% of the whole body standard and less than 3% of the organ dose standard. Episodic releases have made a substantial contribution to the total potential radiation dose only for a few years during test programs of the 1950s and early 1960s. The frequency and size of episodic releases has declined since that time.

SUMMARY

In December 1988, the Idaho National Engineering Laboratory (INEL) Historical Dose Evaluation Task Group was chartered in response to inquiries concerning possible radiological consequences to the public from past projects and facility operations at the INEL. The purpose of this evaluation is to provide estimates of potential offsite radiation doses encompassing the entire operating history of the INEL and using a consistent methodology for all years.

This document describes the methodology and summarizes the results of the offsite dose evaluation. The results presented in Volume 1 are based on an extensive review of many sources of information and a complex series of calculations using meteorological and dosimetric models. Volume 2 contains technical appendices that provide the data and assumptions used, a description of the models and calculations, a listing of results, and citations for reference material.

The evaluation described in this report provides a realistic estimate of potential radiation doses to a hypothetical individual who may have resided at an offsite location with the highest concentration of airborne radionuclides near the INEL. Radiation doses to an adult, child, and infant are estimated. This differs from the type of assessment that attempts to reconstruct actual doses to particular individuals or segments of the population. The location, lifestyle, and habits of this hypothetical individual have been chosen so that it is unlikely that members of the public would have received doses larger than those reported here, and doses to most individuals near the INEL Site would have been much smaller.

The principal route by which radioactivity released on the INEL can reach offsite locations is through the airborne pathway. While potential ground water, surface water, and biotic pathways exist, they are not significant contributors to offsite radiation dose. Consequently, this report concentrates on estimates of doses resulting from radioactivity releases to the atmosphere that are subsequently transported offsite through airborne pathways.

For the purposes of this report, atmospheric releases were categorized as either operational or episodic. Operational releases are continuous, somewhat uniform releases occurring over a year, or over a portion of a year that spans a variety of meteorological conditions. Releases of this type are treated using annual average meteorological conditions. Episodic releases took place over a short period of time, typically a few hours. Some tests as long as several months were modeled as episodic releases. The episodic releases are treated as distinct events using the actual meteorological conditions measured at the time of release. Episodic releases include a variety of tests and experiments as well as a few accidents. It is important to note that most releases were not accidents, but they were short–term releases conducted under predetermined test constraints and specified meteorological conditions.

The best available INEL data were used to estimate the releases of radioactivity to the atmosphere from INEL facilities. The primary source of information on operational releases was the measurement of stack effluents as reported in the Radioactive Waste Management Information System data base. Estimates of earlier releases required supplemental calculations and information from technical reports to determine the radionuclide composition of the effluent. Estimates of episodic releases were obtained from a large number of technical reports on individual facilities and test programs.

The total amount of radioactivity associated with annual operational releases at the INEL was the largest in the late 1950s and early 1960s and has decreased dramatically since that time. The largest amount of radioactivity released in any year during the last decade was approximately one–tenth of that released in 1961. The reduction is a result of programmatic and operational changes, improved effluent control equipment,

and managerial commitment to keep releases as low as reasonably achievable.

A total of 108 individual episodic releases were considered in this evaluation, and detailed dose calculations were done for the 54 most important releases. Most of the important short–term releases were associated with reactor fuel development tests conducted before 1961.

Calculation of atmospheric transport to locations off the INEL Site was done by the Field Research Division of the National Oceanic and Atmospheric Administration using the MESODIF computer code. MESODIF has been extensively tested over many years using data from the Snake River Plain and has been shown to give reliable estimates of atmospheric dispersion. Data for dispersion calculations were taken from the records of the INEL network of meteorological stations, which constitutes one of the best meteorological data bases for a large site.

Radiological assessment calculations were made for four modes of exposure for each age group: (1) external exposure from immersion in contaminated air, (2) external exposure from radioactivity deposited on the ground, (3) internal exposure from inhalation of contaminated air, and (4) internal exposure from ingestion of contaminated agricultural products. Effective dose equivalent for the whole body was calculated along with organ dose commitments to the skin and to five organs: (1) bone surface, (2) red marrow, (3) lung, (4) liver, and (5) thyroid.

Estimated radiation doses were within applicable radiation protection standards in effect at that time for every year of INEL operation. In general, potential offsite doses from INEL activities have been small. The largest radiation doses were calculated for an infant in 1956, when the effective dose equivalent from operational and episodic releases was estimated to be 61 mrem and the thyroid dose equivalent was estimated to be 1350 mrem. Doses from operational and episodic releases during the late 1950s may have been as high as 9% of the applicable whole body dose standard in effect at that time and 90% of the standard for doses to any single organ of the body. In addition, the trend in recent years has been toward smaller doses as total INEL Site releases have declined. For the more restrictive standards in force from 1985 to 1989, offsite radiation doses to a hypothetical individual since 1985 have been about 1% of the whole body dose standard and less than 3% of the organ dose standard. Episodic releases have made a substantial contribution to the total potential radiation dose only for a few years during the test program of the 1950s and 1960s.

Radiation doses from airborne releases over the operating history of the INEL were small when compared to doses from natural background radiation. Doses since the early 1970s were small even when compared to the variability in natural background from year to year and location to location within Southeastern Idaho.

FOREWORD

The Idaho National Engineering Laboratory (INEL) Historical Dose Evaluation Task Group was chartered by the U. S. Department of Energy Idaho Operations Office (DOE–ID) to provide a reasonable but bounding estimate of the radiation dose to a hypothetical, maximally exposed offsite individual attributable to nuclear activities for each year of INEL Site operation. The compilation of records began in November 1988, and the Task Group was formed in December in response to inquiries concerning past projects and facility operations at the INEL Site. The 10 members met regularly, worked in small groups, and sought information from others about the INEL. The dose estimates presented here use the best available data found in INEL records and reports, the judgment of competent professionals, and regulatory methods of dose evaluation.

The *Draft Idaho National Engineering Laboratory Historical Dose Evaluation* was completed in March 1990 and was submitted to an independent Peer Review Panel, which included nine members with special expertise in dose assessment. One member was designated by the State of Idaho, one from the U. S. Environmental Protection Agency, one from Idaho State University, one from the University of Idaho, one from Colorado State University, and four from the private sector. Two of the members had worked at the INEL and had special knowledge of early releases. Three of the members have been involved in dose reconstruction projects at other DOE facilities. The Chairman, Dr. John Till, also chairs the Technical Steering Panel of the Hanford Environmental Dose Reconstruction Project.

The panel performed a thorough and critical review. They made recommendations for improving the draft report and for future action by DOE. The panel concluded that while the report was a commendable effort and will be very useful, it did not meet the stated objectives and did not provide a defensible assessment of doses. The panel recommended that the stated objectives be revised to reflect what was actually accomplished by the Task Group. Other recommendations for improving the draft report were made in the areas of the completeness of the data on the amount of radioactivity released, the methodology applied in the dose evaluation, quality assurance, references to classified documents, treatment of the ground water pathway, and validation of results. Specifically, the panel pointed out that dose assessments for infants and young children, which did not appear in the draft study, are likely to be higher than those for adults.

Each recommendation of the Peer Review Panel has been addressed as completely as possible within the time and resources available to the Task Group. The draft report was extensively revised in response to those comments. The revision took longer than the Peer Review Panel anticipated because of the difficulty in obtaining and evaluating reports from the Aircraft Nuclear Propulsion Program. In several instances, the Task Group found additional changes that were needed but were not included in the recommendations of the panel.

The report is presented in two volumes. Volume 1 contains the results of the dose evaluation in a format intended for the general public. Volume 2, the technical appendices, provides details of the calculations for those who desire additional information including data, references, assumptions, methods, a sample calculation, and detailed results. The complete set of dose calculation results is included on microfiche in the back of Volume 2 of this report. Documenting methods and assumptions and quality assurance checks have been a large part of the effort in revising the draft report. Those records and all of the references cited in the report are available in the INEL Technical Library or the INEL Public Reading Room.

Members of the Task Group who participated in writing this report were E. W. Chew (Chairman), T. F. Gesell, R. L. Dickson, and C. L. Maggart of the DOE–ID; J. F. Sagendorf and G. E. Start of the National Oceanic and Atmospheric Administration; H. K. Peterson of EG&G

Idaho, Inc.; D. R. Wenzel of Westinghouse Idaho Nuclear Company; and M. J. Case and M. D. Otis of Science Applications International Corporation. These individuals were chosen for their expertise in the fields of radiological dose assessment, atmospheric dispersion, and knowledge of INEL Site operations. T. W. Smith and P. H. Allen of DOE–ID met with the Task Group as the report was being revised. Special thanks go to S. J. Francis who was the technical editor of the report.

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ACRONYMS

AEC	U. S. Atomic Energy Commission
ANP	Aircraft Nuclear Propulsion (Program)
ATR	Advanced Test Reactor
CERT	Controlled Environmental Radioiodine Test
DOE	U. S. Department of Energy
DOE–ID	U.S. Department of Energy Idaho Operations Office
EDE	effective dose equivalent
EPA	U. S. Environmental Protection Agency
ERDA	Energy Research and Development Administration
FAST	Fluorinel Dissolution Process and Fuel Storage (facility)
FPFRT	Fission Product Field Release Test
ICPP	Idaho Chemical Processing Plant
IET	Initial Engine Test
INEL	Idaho National Engineering Laboratory
MTR	Materials Testing Reactor
NOAA	National Oceanic and Atmospheric Administration
NRC	U. S. Nuclear Regulatory Commission
NWCF	New Waste Calcining Facility
QA	quality assurance
RSAC	Radiological Safety Analysis Computer Code
RWMIS	Radioactive Waste Management Information System
SL–1	Stationary Low–Power Reactor No. 1
TRA	Test Reactor Area
USGS	U. S. Geological Survey

ABBREVIATIONS

Ci	curie
d	day
h	hour
km	kilometer
m	meter
mi	mile
mrem	millirem
yr	year

IDAHO NATIONAL ENGINEERING LABORATORY HISTORICAL DOSE EVALUATION

1. INTRODUCTION

1.1 Purpose and Goals

The Idaho National Engineering Laboratory (INEL) Historical Dose Evaluation is a study that estimates the radiation doses potentially received by individuals who lived near the INEL Site boundary for each year of INEL operation from 1952 through 1989. Only radioactive releases to air are examined because this is the principal pathway by which offsite residents have been exposed to radiation from the INEL. All significant known accidents, tests, experiments, and routine operational releases are included in the evaluation.

Radiation dose estimates are made using U. S. Environmental Protection Agency (EPA) and U.S. Nuclear Regulatory Commission (NRC) radiation dose assessment methods that apply generic models to calculating dose to hypothetical individuals. To account for changes in diet and metabolism, doses are calculated for three age groups: adult, child, and infant. The calculations are intended to be conservative but realistic. They do not use a combination of all "worst case" assumptions, but they use best estimates of source terms, meteorological dispersion parameters, and dose conversion factors.

The study uses the dosimetric units of effective dose equivalent (EDE) consistent with current radiation protection practice. EDEs and radiation doses to specific organs were calculated, using currently accepted values of dose conversion factors, for comparison to whole body and critical organ standards in effect during past years.

The INEL Historical Dose Evaluation is a preliminary study that investigates airborne releases of radioactivity from the INEL Site, particularly those in the early years of Site operations, and estimates the potential offsite impacts. The study does not attempt to reconstruct actual doses to particular individuals or segments of the population. The study does not calculate occupational doses to INEL workers because worker doses are being addressed in another study. Nor is the study intended to address releases of potentially hazardous chemical pollutants. As a preliminary study, the INEL Historical Dose Evaluation depends primarily on historical data available in published reports. Verification of the calculated doses by extensive new interpretation of existing monitoring data or by new measurements of environmental samples is beyond the scope of the present study. Studies of the surrounding population to determine historical age distributions, locations of residence, lifestyle, dietary habits, and agricultural practices are also beyond the scope of this study.

The goal of the INEL Historical Dose Evaluation has been to provide a scientific basis for determining whether a detailed reconstruction of radiation doses to actual populations living in areas adjacent to the INEL Site is warranted. The report provides a compilation of references to historical material on early INEL Site operation. It also brings together information necessary to evaluate the importance of early INEL Site releases in comparison to standards and guidelines.

The INEL Historical Dose Evaluation provides the basic data for informed public involvement. The U. S. Department of Energy (DOE) has entered into an agreement with the State of Idaho for several health–related projects, including a review of the models used to estimate potential radiation doses to the public from INEL operations. Under this agreement, the State has established an Advisory Panel to conduct the review and make recommendations to the State. The DOE has recently notified the State of Idaho that it will seek funding for the Centers for Disease Control to perform an INEL dose reconstruction under an existing Memorandum of Understanding, but the dose reconstruction will be

independent from DOE direction. The Centers for Disease Control study will have full public involvement.

1.2 Past Assessments

Although dose assessments were performed for a variety of purposes throughout the history of the INEL Site, a comprehensive evaluation of the offsite doses, using a consistent methodology and encompassing the entire operating history of the INEL, had not been performed.

Other assessments of potential offsite doses were done for the INEL and are available to the public. A compilation of radionuclides released from INEL Site operations, an evaluation of potential doses to a member of the general public at a nearby offsite location, and related environmental monitoring data were published annually since 1973 (Hoff et al., 1990). During the years from 1959 (HEW, 1960) through 1973, INEL Site environmental data were published in the U. S. Department of Health, Education, and Welfare report *Radiological Health Data.* Before this period, data on potential doses or radioactivity releases for individual programs and tests were published in U. S. Atomic Energy Commission (AEC) reports.

These and other assessments done for the INEL used the regulatory guidance and assessment methodologies available at the time. However, the state of the art in radiological assessment has evolved over the 40 years of INEL Site operation, and regulatory standards have changed. Even some of the fundamental definitions of radiation dose have undergone changes during this period. Consequently, it is difficult to compare the results of past assessments on a consistent and common basis. The evaluation performed for this report was designed to provide estimates of offsite doses calculated using current regulatory assessment methods in a consistent manner over the entire operating history of the INEL Site.

1.3 Background of the INEL

The National Reactor Testing Station was established by the Federal Government in 1949 to provide a site where various kinds of nuclear reactors and support facilities could be built and tested. The name was changed in August 1974 to the Idaho National Engineering Laboratory to better reflect the broader mission of the INEL Site. Over the years, the INEL Site has been operated by the AEC, the Energy Research and Development Administration (ERDA), and it is currently operated by DOE.

The INEL Site is located on the Upper Snake River Plain in Southeastern Idaho and includes an area of 2300 km^2 (890 mi^2). It is isolated from large population centers. The INEL Site boundaries are 35 km (22 mi) west of Idaho Falls, 37 km (23 mi) northwest of Blackfoot, 71 km (44 mi) northwest of Pocatello, and 11 km (7 mi) east of Arco. With a current population of 1200, Arco is the largest boundary community in the area of the INEL Site. Smaller boundary communities around the INEL include Atomic City to the south, Howe to the west, and Terreton and Mud Lake to the northeast. Approximately 120,600 people reside within a radius of 80 km (50 mi) of the INEL Site's operational center but none within 16 km (10 mi) of that center. A complete description of the environmental and demographic setting of the INEL is available in the *INEL Environmental Site Characterization Report* (EG&G, 1984).

The first nuclear reactor on the INEL Site was the Experimental Breeder Reactor No. 1, which achieved initial criticality in 1951. The first major airborne releases of radioactivity occurred in 1952 when the Materials Testing Reactor (MTR) began operation. The Idaho Chemical Processing Plant (ICPP), where uranium is recovered from spent fuel, has been in operation since 1953. Since the INEL Site was established, 52 reactors were built and tested. Thirteen of them were operating or operable in 1988. Figure 1–1 is a map of the INEL Site showing major facility locations. Table 1–1 is a tabulation of facilities and programs and their associated acronyms. Figure 1–2

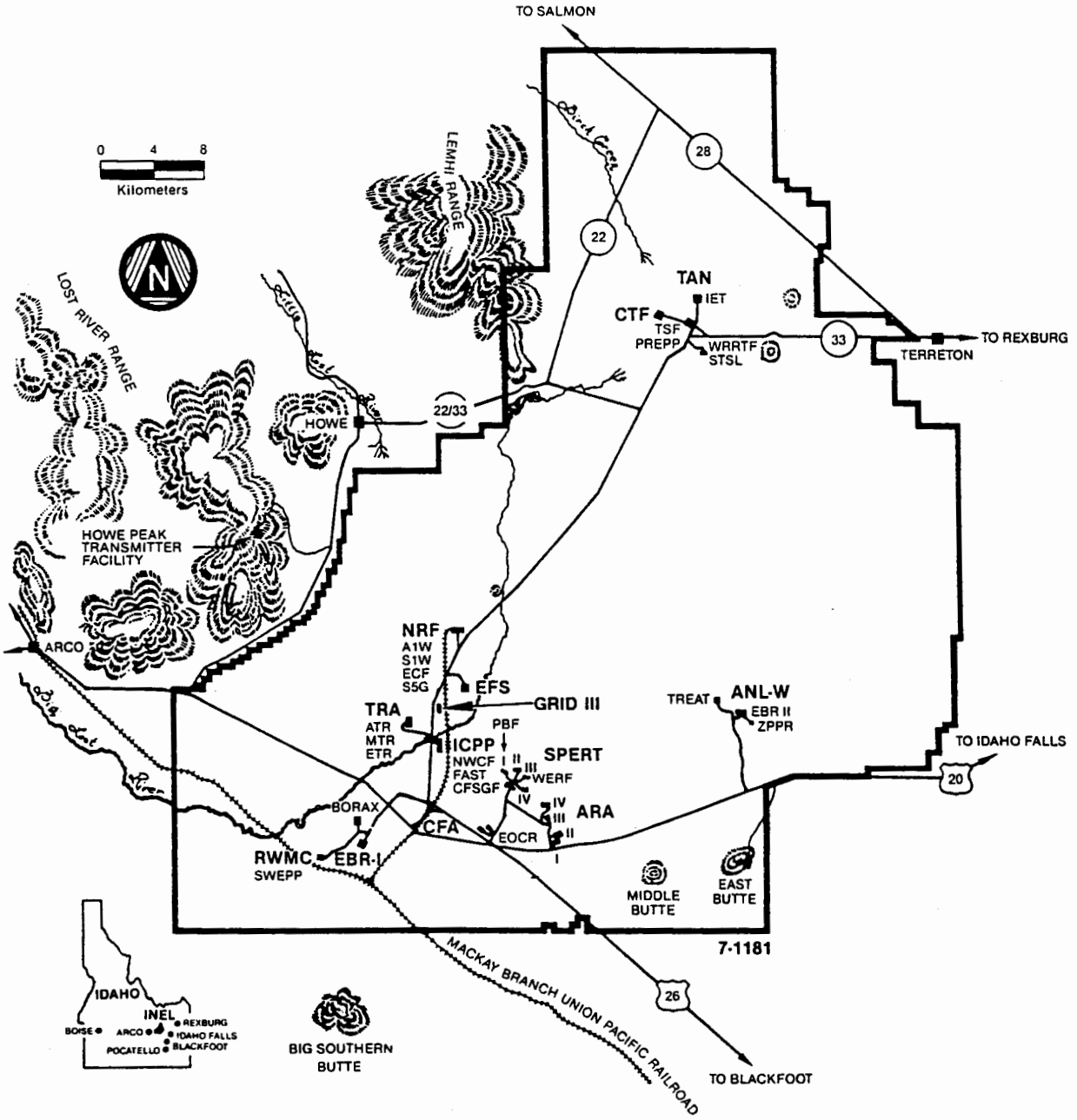


Figure 1–1. INEL Site facility locations.

Table 1–1. Facilities and programs at the INEL

Name	Abbreviation	Operating Contractor[a]
Reactors Operating or Operable as of December 1984		
Advanced Reactivity Measurement Facility No. 1[b]	ARMF–1	EG&G
Advanced Test Reactor	ATR	EG&G
Advanced Test Reactor Critical	ATRC	EG&G
Argonne Fast Source Reactor	AFSR	ANL
Coupled Fast Reactivity Measurement Facility[b]	CFRMF	EG&G
Experimental Breeder Reactor No. 2	EBR–II	ANL
Large Ship Reactor "A"	A1W–(A)	WEC
Large Ship Reactor "B"	A1W–(B)	WEC
Natural Circulation Reactor	S5G	WEC
Submarine Thermal Reactor	S1W (STR)	WEC
Transient Reactor Test Facility	TREAT	ANL
Neutron Radiography Facility	NRAD	ANL
Zero Power Physics Reactor[b]	ZPPR	ANL
Reactors Dismantled, Transferred, or in Standby Status		
Boiling Water Reactor No. 1	BORAX–I	ANL
Boiling Water Reactor No. 2	BORAX–II	ANL
Boiling Water Reactor No. 3	BORAX–III	ANL
Boiling Water Reactor No. 4	BORAX–IV	ANL
Boiling Water Reactor No. 5	BORAX–V	ANL
Engineering Test Reactor	ETR	EG&G
Engineering Test Reactor Critical	ETRC	EG&G
Experimental Breeder Reactor No. 1	EBR–I	ANL
Experimental Organic Cooled Reactor (Mothballed before startup)	EOCR	PPCo
Gas Cooled Reactor Experiment	GCRE	—
Loss–of–Fluid Test Facility	LOFT	EG&G
Materials Testing Reactor	MTR	PPCo & INC
Mobile Low Power Reactor No. 1	ML–1	—
Organic Moderated Reactor Experiment	OMRE	AI
Power Burst Facility	PBF	EG&G
Special Power Excursion Reactor Test No. 1	SPERT–I	PPCo
Special Power Excursion Reactor Test No. 2	SPERT–II	PPCo & INC
Special Power Excursion Reactor Test No. 3	SPERT–III	PPCo & INC
Special Power Excursion Reactor Test No. 4	SPERT–IV	PPCo & INC
Spherical Cavity Reactor Critical Experiment	SCRCE	ANC
Stationary Low–Power Reactor No. 1	SL–1	—
Zero Power Reactor No. 3[b]	ZPR–III	ANL
Other Facilities in Use		
Argonne National Laboratory–West	ANL–W	ANL
Auxiliary Reactor Area	ARA	EG&G
Central Facilities Area	CFA	EG&G
Chemical Engineering Laboratory	CEL	EG&G
Coal–Fired Steam Generating Facility	CFSGF	WINCO
Computer Science Center (Idaho Falls)	CSC	EG&G
Contained Test Facility	CTF	EG&G
Expended Core Facility	ECF	WEC
Experimental Field Station	EFS	DOE–ID
Fluorinel and Fuel Storage Facility	FAST	WINCO
Fuel Manufacturing Facility	FMF	ANL
Hot Cell Facility (TRA)	—	EG&G
Hot Fuel Examination Facilities	HFEF	ANL
Other Facilities in Use (continued)		
Hot Shop Facilities (TAN)	—	EG&G
Idaho Chemical Processing Plant	ICPP	WINCO
INEL Research Center (Idaho Falls)	IRC	EG&G
Naval Reactors Facility	NRF	WEC
New Waste Calcining Facility	NWCF	WINCO
Process Experimental Pilot Plant	PREPP	EG&G
Radiation Measurements Laboratory	RML	EG&G
Radioactive Waste Management Complex	RWMC	EG&G
Radiological and Environmental Sciences Laboratory	RESL/ID	DOE–ID
Reactor Training Facility	RTF	EG&G
Remote Analytical Laboratory	RAL	WINCO
Semiscale Test Support Laboratory	STSL	EG&G
Standards Calibration Laboratory (CF–698)	—	EG&G
Stored Waste Examination Pilot Plant	SWEPP	EG&G
Technical Services Center (CF–688, –699)	TSC	EG&G
Technical Service Facility	TSF	EG&G
Technical Support Buildings A&B (Idaho Falls)	TSA;TSB	EG&G
Test Area North	TAN	EG&G
Test Reactor Area	TRA	EG&G
Waste Experimental Reduction Facility	WERF	EG&G
Water Reactor Research Test Facility	WRRTF	EG&G
Willow Creek Building (Idaho Falls)	WCB	EG&G
Facilities Not Presently in Use		
Initial Engine Test Facility	IET	EG&G
Field Engineering Test Facility	FET	EG&G
Waste Calcining Facility	WCF	WINCO
Past Programs at the INEL		
Aircraft Nuclear Propulsion	ANP	—
Controlled Environmental Radioiodine Test	CERT	—
Fission Product Field Release Test	FPFRT	—
SNAP 10A Transient	SNAPTRAN	—
Major Programs at the INEL		
Chemical Processing Program	—	WINCO
Liquid Metal Fast Breeder Reactor Program	—	ANL
Naval Propulsion Reactors Program	—	WEC
Radioactive Waste Management Program	—	EG&G
Reactor Materials Testing Program	—	EG&G
Specific Manufacturing Capability	—	R–INEL
Water Reactor Safety Program	—	EG&G

a. Operating contractor acronyms: Atomic International (AI), Aerojet Nuclear Company (ANC), Argonne National Laboratory (ANL), EG&G Idaho, Inc. (EG&G), Exxon Nuclear Idaho Company, Inc. (ENICO), Idaho Nuclear Corporation (INC), Phillips Petroleum Company (PPCo), Rockwell–INEL (R–INEL), Westinghouse Electric Corporation (WEC), Westinghouse Idaho Nuclear Company (WINCO).

b. Zero or low power reactor.

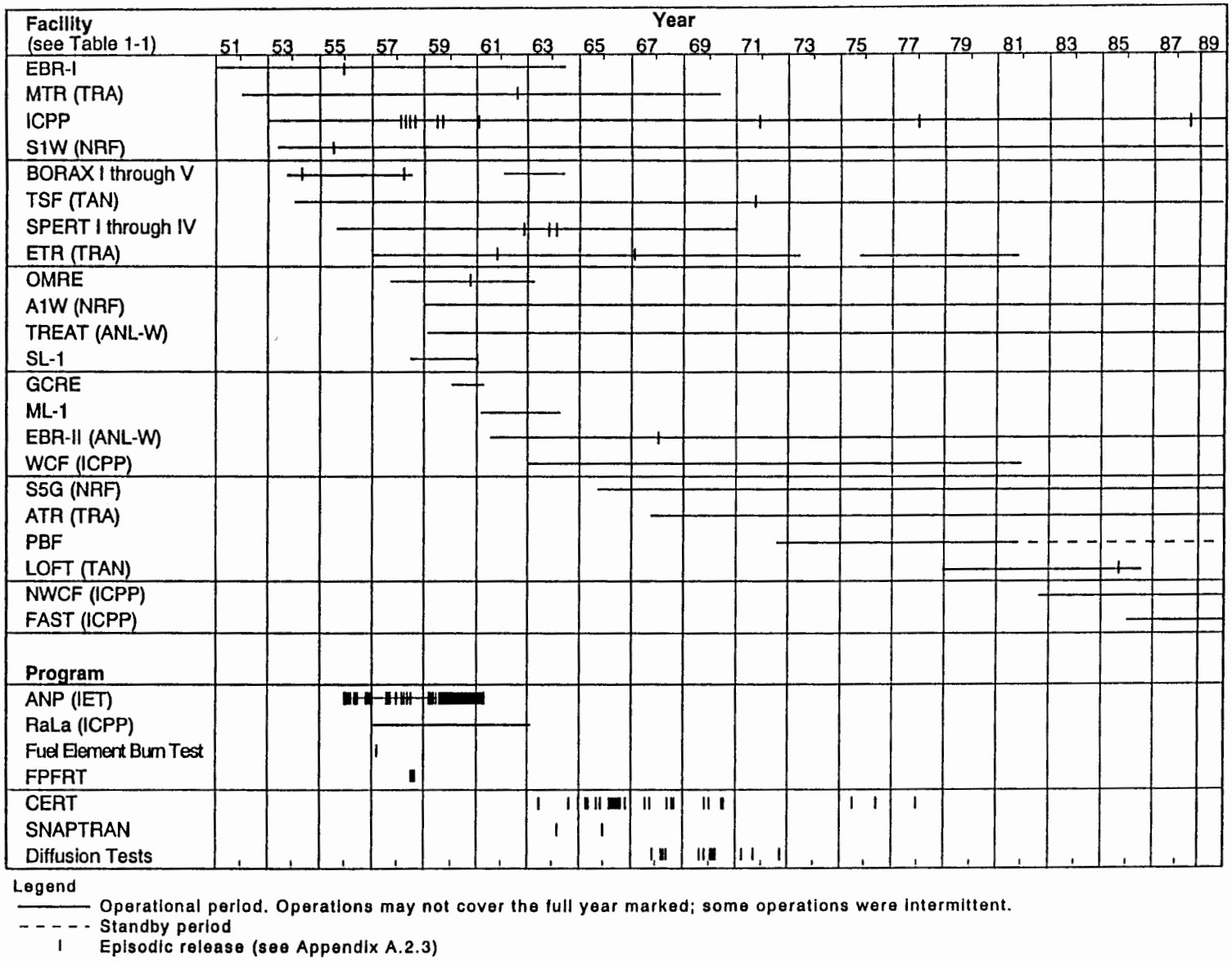


Figure 1–2. Chronology of selected INEL facilities and programs.

displays the chronology of selected INEL facilities and programs. Further information on these and other INEL facilities and programs can be found in a variety of documents including the *Final Environmental Impact Statement, Waste Management Operations* (ERDA, 1977a) and the latest INEL informational brochure (INEL, 1988). The first report is a comprehensive description of waste management activities as they existed at the INEL in 1974. It includes a history of the activities of the INEL, releases of radioactive material to the environment, and an extensive list of references. The second is a summary of current INEL operations and a brief description of formerly used facilities.

1.4 Potential Exposure Pathways

A number of potential exposure pathways exist by which radioactive materials from INEL Site operations could be transported to offsite environments. These exposure pathways are illustrated in Figure 1–3. The airborne pathway is the principal pathway by which radioactive materials released on the INEL can reach an offsite member of the public. Inhalation and direct radiation from the airborne pathway are two of the four pathways evaluated in this report. The indirect pathway from airborne materials to meat, milk, and vegetables through ingestion is the third pathway included in this report. The fourth major pathway evaluated is the ground deposition pathway in which airborne materials deposited on the soil surface produce direct radiation exposure.

Liquidborne radioactive effluents have not, to this time, produced measurable exposure to an offsite member of the public. No appreciable amount of surface water flows from the INEL Site to offsite areas. The Big Lost River, Little Lost River, and Birch Creek are intermittent streams that flow onto the INEL and sink into the porous soils and basalts of the INEL Site, ultimately recharging the Snake River Plain Aquifer. There are no liquid discharges to these streams from INEL operations. In the past, liquid radioactive materials have been disposed of directly to the Snake River Plain Aquifer through injection wells. The practice was discontinued in 1984. Surface disposal ponds are used to dispose of low–level radioactive liquids. Water from these ponds percolates through the soil and basalt to the Snake River Plain Aquifer. Some of the radionuclides are sorbed on the soil column before they reach the aquifer (e.g., cesium and plutonium). Others, including tritium, migrate rapidly with the water through the soil column. Other radionuclides, such as strontium, are retarded by the soil column, but they may reach the aquifer from the disposal ponds.

The quantities of liquid radioactive materials released at the INEL were tabulated through 1974 in the *Final Environmental Impact Statement, Waste Management Operations* (ERDA, 1977a). Quantities released in recent years have declined, as shown in Table 1–2. The extent of radioactivity in the Snake River Plain Aquifer has been reported by the U. S. Geological Survey (USGS) for many years (e.g., Barraclough et al., 1967). The distribution of tritium and strontium–90 in the aquifer was shown in ERDA (1977a). Some of the more recent USGS reports show the distribution of radionuclides and chemicals in the aquifer (Mann and Cecil, 1990; Pittman et al., 1988; Knobel and Mann, 1988; Mann et al., 1988; Mann and Knobel, 1988; Mann and Knobel, 1987). These reports specifically cover radionuclides (tritium, cobalt–60, strontium–90, iodine–129, cesium–137, and plutonium isotopes); chemical constituents (sodium, chromium, chloride, sulfate, nitrate, nine trace metals, and purgeable organic compounds); and indicator parameters such as specific conductance, temperature, and pH.

The radionuclide that has moved the farthest in the aquifer from its point of origin on the INEL is chlorine–36. A sensitive technique, accelerator mass spectrometry, was used to identify chlorine–36 in wells at the INEL Site boundary and possibly in two offsite monitoring wells in 1990. However, the offsite concentrations are one million times smaller than the EPA community drinking water standard for man made

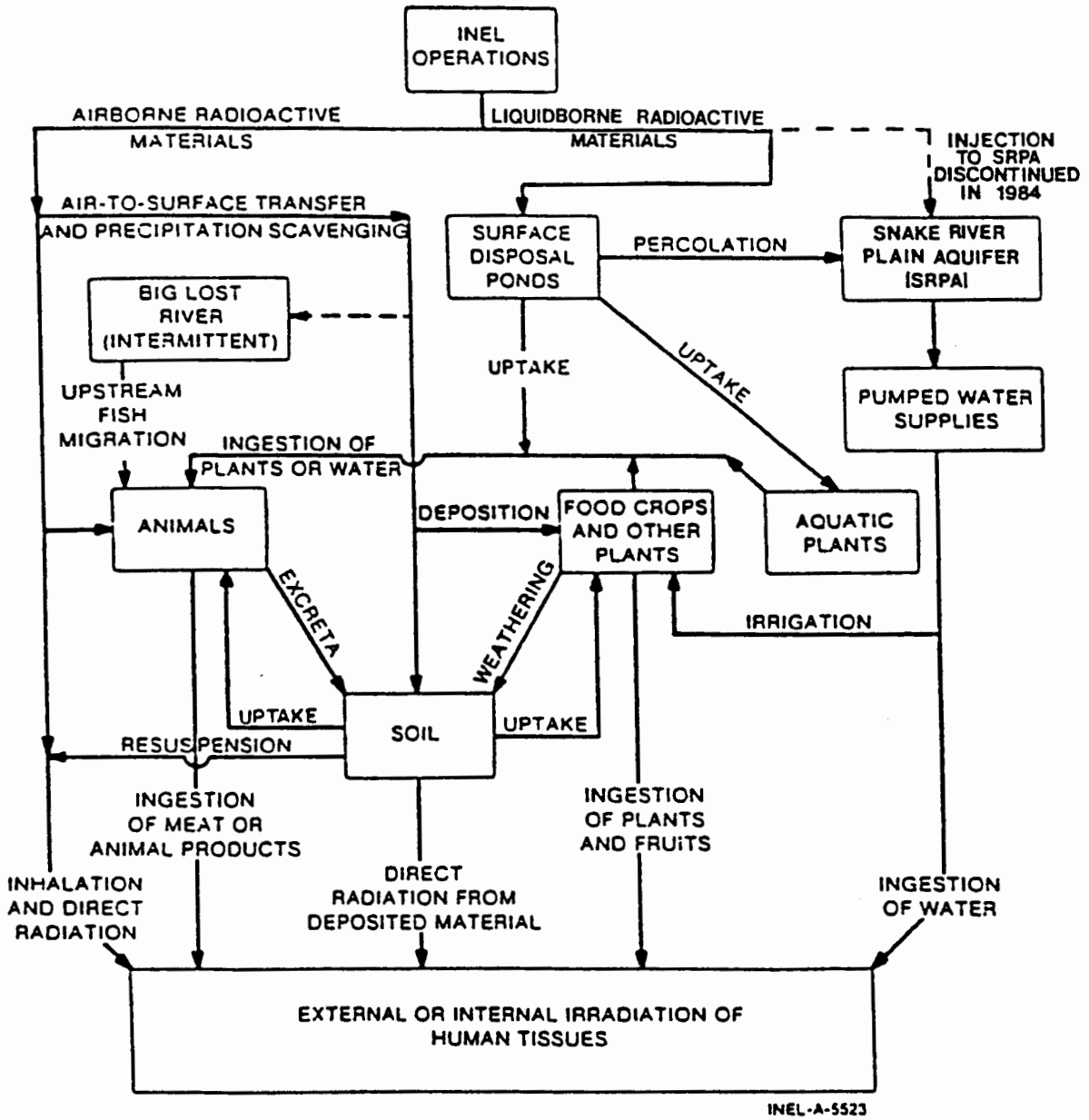


Figure 1–3. Possible exposure pathways of the INEL Site radioactive materials to the public.

radionuclides. Chlorine–36 serves as a useful and sensitive ground water tracer at the INEL, but it does not pose a health hazard.

The radionuclide that is documented as moving the next farthest in the aquifer is tritium. Tritium is an isotope of hydrogen and is incorporated into water molecules in the aquifer. Its distribution has been followed for many years. Its distribution in the aquifer in 1988 is shown in Figure 1–4. From 1983 through 1985, tritium was measured in wells on the INEL Site boundary (Hoff et al., 1990). However, tritium attributable to the INEL operations has not been detected in wells beyond the INEL Site boundary. The minimum detectable concentration for tritium is one–fiftieth of the EPA community drinking water standard. Thus, there has been no significant dose to an offsite member of the public through the Snake River Plain Aquifer pathway. Predicting the movement of radionuclides in the aquifer for

Table 1–2. INEL radioactive liquid waste discharges

Year	Curies	Year	Curies	Year	Curies
1952	63	1965	763	1978	1572
1953	232	1966	1195	1979	1722
1954	1152	1967	1907	1980	376
1955	1277	1968	1331	1981	577
1956	869	1969	3456	1982	809
1957	1256	1970	4575	1983	675
1958	3447	1971	2971	1984	191
1959	4863	1972	1711	1985	667
1960	3508	1973	1557	1986	346
1961	4519	1974	2248	1987	357
1962	1738	1975	2227	1988	270
1963	2434	1976	3325	1989	137
1964	2676	1977	2028		
				TOTAL	75,717

Sources: ERDA (1977a, p. III–61); Litteer and Reagan (1990); Batchelder and Cassidy (1984). Amounts differ slightly depending upon the reference used.

long time periods is beyond the scope of this study. Previous USGS reports have documented studies about long–term movement of radionuclides in the aquifer (Robertson, 1974; Lewis and Goldstein, 1982). Lewis and Goldstein (1982) found that the model used by Robertson tended to predict concentrations for 1980 that were higher than those actually observed. Because there have been no significant doses from this pathway and future doses are not expected to exceed EPA drinking water standards, the ground water pathway will not be evaluated further in this report.

Two other pathways will not be treated in this report: precipitation scavenging of airborne radioactive materials and resuspension of contaminated soil. Precipitation scavenging, the washing of radioactive materials from the air by rain, may be important in the local area where it occurs. Because of the low rainfall in the INEL area and the tendency for much of the rain or snow to fall as scattered showers, the precipitation scavenging pathway has not been included in this report. The resuspension of contaminated surface soil is not an important pathway for this analysis. Analyses of continuous air samples from the INEL Site boundary have only rarely detected radionuclides that might have come from contaminated soils. Analyses of air samples from the Radioactive Waste Management Complex, one of the INEL facilities with the greatest potential for resuspension, indicate that doses at the INEL Site boundary would be less than the other doses evaluated in this report (Hoff et al., 1990).

Some potential biotic pathways (i.e., animals and vegetation) do exist at the INEL. Much research has been done on these pathways and is reported in peer–reviewed literature. Reynolds and Markham (1987), *Publications of the Idaho National Engineering Laboratory Radioecology and Ecology Program: 1974–1986*, lists many of these reports; updates are available. The most important biotic pathway is through game animals that can assimilate some radioactivity onsite and then migrate offsite. However, the probability of a hunter shooting one of these animals shortly after the animal leaves the INEL is small.

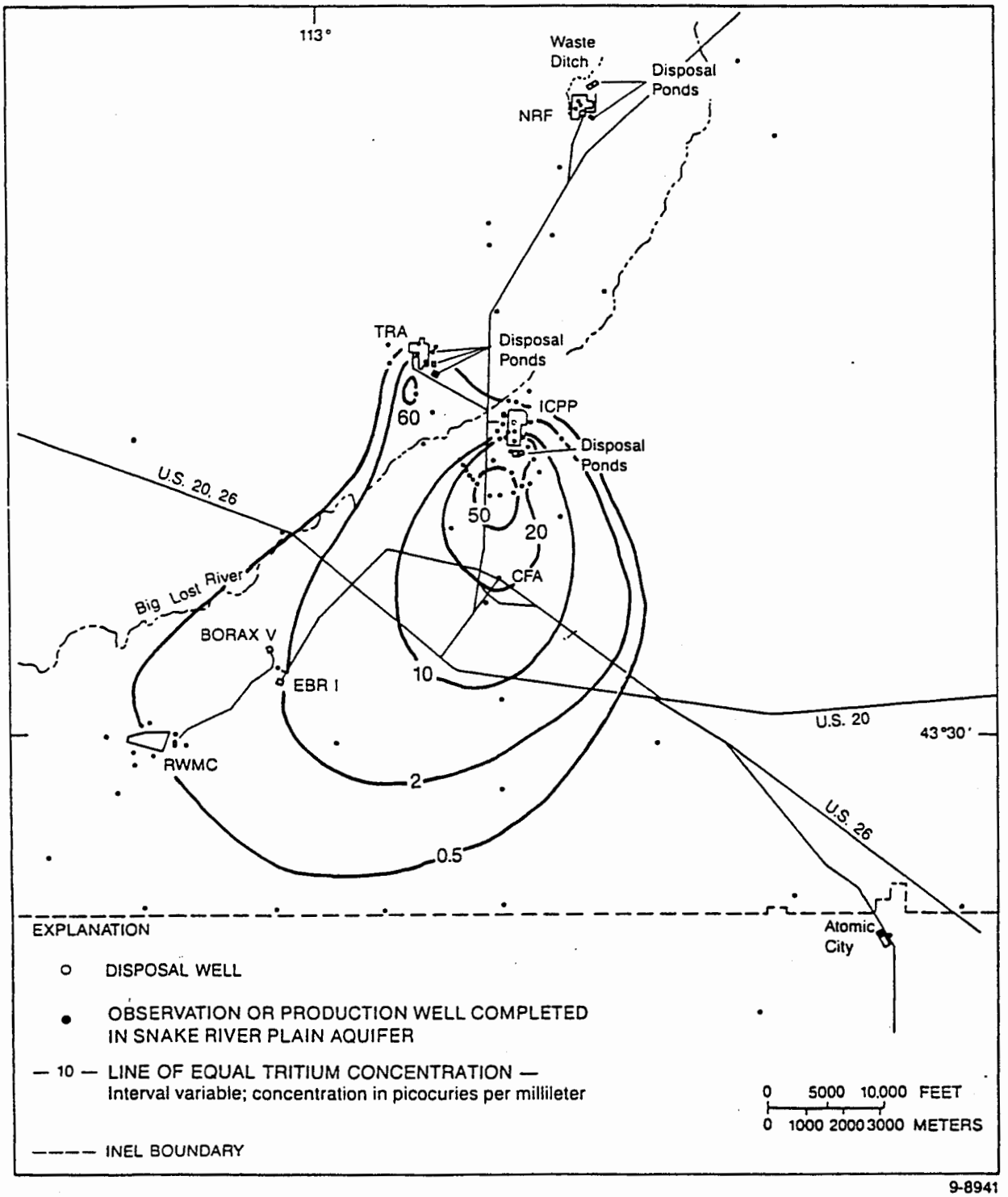


Figure 1–4. Distribution of tritium in the Snake River Plain Aquifer in the southcentral part of the INEL, 1988.

The potential for radiation dose to people offsite through game animals is discussed in Hoff et al. (1990) and in numerous research reports. Doses, although unlikely, might be as high as 10 mrem.

In conclusion, the airborne pathway is the principal pathway by which radioactive materials released from INEL activities can reach members of the public offsite. Consequently, this report concentrates on estimating doses from the four major airborne pathways: inhalation, immersion, ground deposition, and ingestion of agricultural products.

1.5 Organization of the Document

This document describes the methodology and summarizes the results of the offsite dose evaluation. The results presented in Volume 1 are based on an extensive review of many sources of information and a complex series of calculations using meteorological and dosimetric models. Volume 2 contains technical appendices that provide the data and assumptions used, a description of models and calculations, a listing of all results, and citations for reference material.

The dose evaluation process begins with estimates of the amounts of radioactivity released. Airborne transport to the INEL Site boundary and surrounding areas is then modeled. Finally, radiation doses are calculated based on the resulting air concentrations. Estimates of the amount of radioactivity released are discussed and summarized in Section 2, and airborne transport is discussed in Section 3 of this report. Section 4 presents results in terms of doses to a hypothetical individual and a comparison of results to applicable regulatory standards and the natural radiation background of Southeastern Idaho. Section 5 describes quality assurance activities performed while preparing this report, Section 6 contains a glossary of terms used in this evaluation, and Section 7 presents a list of references cited in this volume of the report.

2. QUANTITY OF RADIONUCLIDES RELEASED

Determining the amount of radioactivity released is the first step in calculating dose. A detailed description of how these quantities, or source terms, were determined is presented in Appendix A.

Releases were categorized as either operational or episodic for purposes of this report. The task of determining the amount of radioactivity released was divided into two parts because of the differing requirements of the atmospheric dispersion calculations for the two types of releases. Operational releases are continuous, somewhat uniform releases occurring over a year, or over a portion of a year that spans a variety of meteorological conditions. Releases of this type can be treated using annual average meteorological conditions. Episodic releases are the tests, experiments, and accidents that took place over a short period of time (typically a few hours). Several of the tests during the Aircraft Nuclear Propulsion (ANP) Program were performed over several weeks to several months and were also treated as episodic releases. It is important to note that most of these episodic releases are not due to accidents. Most were short–term releases that resulted from experiments and tests conducted under predetermined test constraints and specified meteorological conditions. For example, the Controlled Environmental Radioiodine Test (CERT) series involved intentional releases of very small and controlled amounts of tracers in studies designed to evaluate environmental behavior of radionuclides under conditions typical on the INEL Site. Episodic releases are treated as distinct events. They are not averaged over a year's time, but they are modeled with the actual meteorological conditions measured at the time of the test.

The best and most comprehensive available data on releases of radioactivity to the atmosphere from INEL facilities have been used. The primary source of information on operational releases was the measurement of stack effluents as reported in the Radioactive Waste Management Information System (RWMIS) data base (Litteer and Reagan, 1990). The data in RWMIS are generally most complete and reliable for the years since this data base was established in 1971. Data for the years from 1962 to 1970 are less complete and reflect less sophisticated monitoring instrumentation and record keeping systems. Data from 1952 to 1961 do not identify the specific mixture of radionuclides released; therefore, additional assumptions and calculations were required in order to determine the radionuclide composition of the effluent and perform the dose assessment.

Not all atmospheric releases are included in the RWMIS data base. Many short–term or episodic releases were reported in separate documents and reports. These releases include a wide variety of tests and experiments and the few accidental releases that have occurred at the INEL Site. A comprehensive review of these reports was conducted to locate any information available on amounts of radioactivity released, radionuclide mixture, and circumstances of release for each episode. A list of references used for episodic releases appears in Appendix A of this report. In many cases, supplementary calculations were required to develop sufficient information for a radiological dose assessment when details concerning the source term were not available in technical reports.

Supplementary calculations to support the development of the operational and episodic source terms fall into two major categories. The first are those calculations required to adjust the RWMIS data to account for radioactive decay, ingrowth of daughter nuclides, and changes in monitoring instrumentation. The second are those calculations used to estimate the radionuclide mixture for instances where only the total amount of radioactivity released is recorded. For operational releases reported in the second category, the radionuclide mixture was assumed to be similar to the composition measured at other times under similar operating conditions. For episodic releases, calculations were performed using portions of the Radiological Safety Analysis Computer Program (RSAC–4), which can

calculate the inventory of radionuclides in a nuclear reactor based on characteristics of its fuel and operating history. Both types of calculations require familiarity with the operation of various INEL facilities, extensive use of available reports, and careful application of scientific and engineering judgment. The results are summarized in the following sections. Further detailed information can be found in Appendix A.

2.1 Operational Releases of Radionuclides

The total amount of radioactivity associated with operational releases for each year is shown in Figure 2–1. These values do not include the amount of radioactivity associated with episodic releases discussed in Section 2.2. As can be seen from the figure, releases in recent years have been lower than releases in the early years. The largest amount of radioactivity released from the INEL over the period of 1 year occurred in 1961 when 1,500,000 Ci were released. The largest amount of radioactivity released in any year during the last 10 years has been about one–tenth of that released in 1961. The reduction of the amount of radioactivity released is a result of programmatic and operational changes, improved effluent control equipment, and managerial commitment to keep releases at the INEL as low as reasonably achievable.

The total amounts of radioactivity shown in Figure 2–1 are not directly proportional to the dose to an individual offsite. Each radionuclide behaves differently and results in a different radiation dose for each curie released. Further evaluations based on the specific radionuclides in the release and meteorological conditions are needed to determine potential doses. This detailed information appears in Appendix A. The quantities shown in Figure 2–1 are the amount of radioactivity released at the individual INEL facilities and were not corrected for the

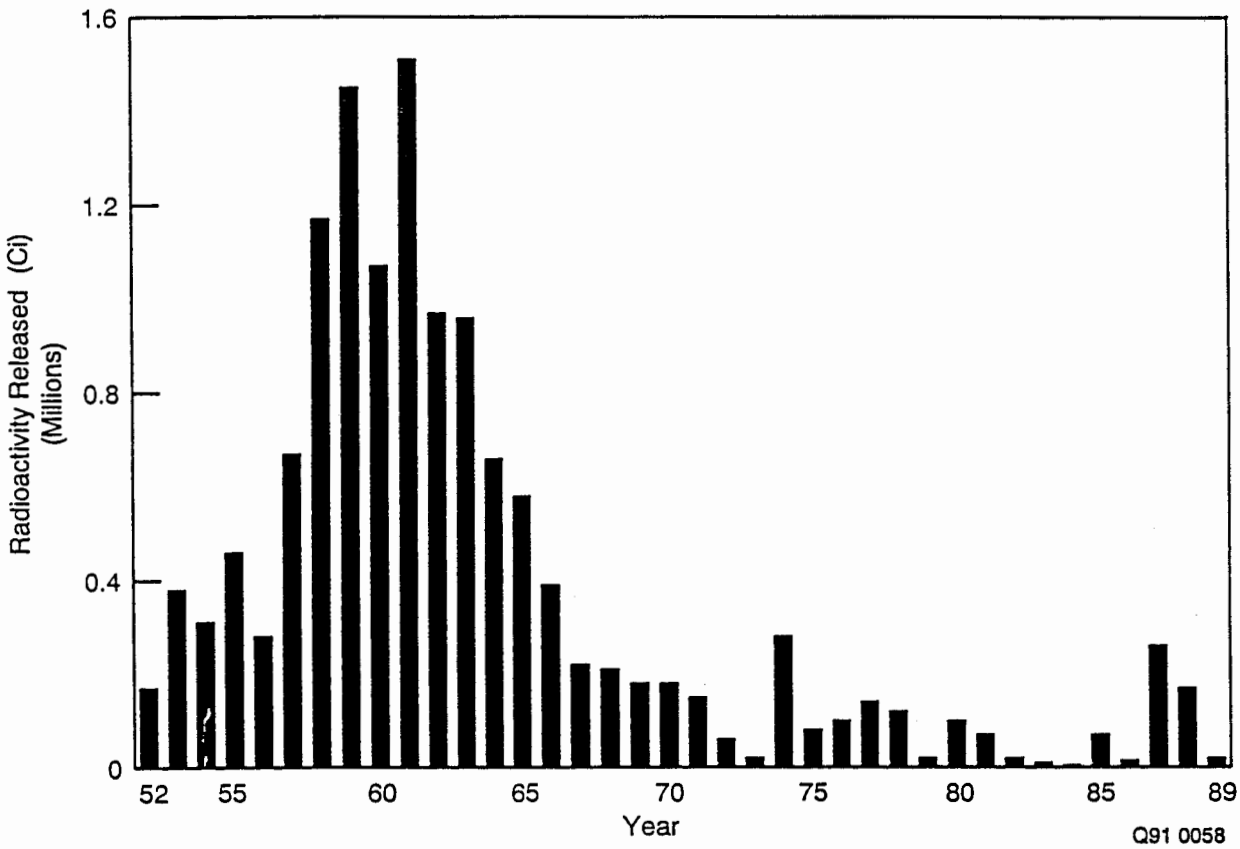


Figure 2–1. Annual quantity of radionuclides released at the INEL Site, operational releases.

radioactive decay that takes place as the radionuclides travel to the INEL Site boundary. Radioactive decay is accounted for in the dose calculation for operational releases.

2.2 Episodic Releases of Radionuclides

The amounts of radioactivity released during the episodic events are shown in Table 2–1. The 23 episodic releases (including one release divided into two parts and one release divided into three parts) listed in the table are those that were calculated to contribute an EDE to an exposed individual offsite of 0.1 mrem or greater. The complete list of 108 episodic events considered is shown in Appendix A, Table A–19. The episodic releases are listed from the most recent to the earliest. Again, the trend can be seen for larger and more frequent releases in the earlier years of INEL Site operations.

Table 2–1. Amount of radioactivity released from the INEL Site for selected episodic events[a]

Date	Description	Area (present name)	Total Activity[b] (Ci)
1/3/61	SL–1 Accident	ARA	1,100
3/1–3/30/61	IET #26(B)	IET	3,100
9/7–10/14/60	IET #23 (FEET #2)	IET	1,700
5/14–6/10/60	IET #20	IET	7,500
3/1–4/30/60	IET #19(B)	IET	8,400
1/6–2/7/60	IET #18	IET	14,000
10/16/59	1959 ICPP Criticality	ICPP	37,000
4/24–5/19/59	IET #14	IET	7,500
11/18/58	IET #13 (HTRE No. 3 Excursion)	IET	940
10/29–10/30/58	FECF Filter Break	ICPP	3.9
9/26/58	FPFRT–9	GRID III	10
9/18/58	FPFRT–8	GRID III	100
9/17/58	FPFRT–7	GRID III	90
8/27/58	FPFRT–5	GRID III	140
8/14/58	FPFRT–4	GRID III	9.6
8/6/58	FPFRT–3	GRID III	9.9
8/4/58	FPFRT–2	GRID III	9.3
3/20–4/14/58	IET #11	IET	4,200
3/1–3/6/58	IET #10(B)	IET	140,000
12/20/57–2/25/58	IET #10(A)	IET	130,000
3/20/57	Fuel Element Burn Test, B	GRID III	74
6/29/56	IET #4(C)	IET	150,000
5/24–6/26/56	IET #4(B)	IET	210,000
5/1–5/23/56	IET #4(A)	IET	6,800
2/11–2/24/56	IET #3	IET	46,000
6/18–7/1/55	NRF S1W Engineering Test	NRF	310

a. Episodic releases producing estimated doses less than 0.1 mrem are not included in this table.

b. The activity reported includes the reduction because of the radioactive decay that occurred during transport from the point of release to the INEL Site boundary.

The quantities shown for episodic releases reflect those radionuclides primarily responsible for the radiation doses (see Section A.2.3.1) and the amount of radioactivity that reached the INEL Site boundary. Decay of radioactivity during transport to the INEL Site boundary has been included for many of the releases. As indicated above, the total amount of radioactivity is not directly related to the dose to an offsite individual; but rather, the contributions of the specific radionuclides involved need to be evaluated individually. The detailed breakdown of radionuclides used in the episodic dose evaluations are provided in Appendix A and on microfiche included in Volume 2 of this report. Weather conditions at the time of the release, which affect the dispersion of radionuclides, are also important. A discussion of these factors is presented in the following section and in Appendix B.

3. ATMOSPHERIC DISPERSION

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 (Start and Wendell, 1974). The name MESODIF is derived from the terms mesocale and diffusion to indicate that the dispersion predictions of the model are valid for distances up to about 150 km (95 mi) from the release point. The MESODIF computer code has been extensively tested using data for the Snake River Plain and has been shown to give reliable estimates of atmospheric dispersion over many years (Wendell, 1970; Wendell, 1972; Yanskey et al., 1966; Islitzer and Dumbauld, 1963; Islitzer, 1961; Lewellen et al., 1985; Start et al., 1985; Sagendorf and Fairobent, 1986; Hoff et al., 1987; DOE, 1987). MESODIF has been used in other assessments of annual average offsite radiation doses at the INEL Site since 1973. MESODIF is also used as part of the standard procedure for predicting the consequences of releases at the INEL. A description of the MESODIF computer code and the results of all calculations used in this evaluation are given in Appendix B and in the references cited there.

One key to the ability of the model to correctly calculate atmospheric dispersion is the availability of detailed meteorological data. The INEL is recognized as having one of the best meteorological data bases for a site of its size. An extensive network of meteorological stations has been in operation since 1969. The number and location of the stations has changed over the years. There are currently 26 stations providing meteorological data every 6 minutes for the Snake River Plain. Before 1969, archival hourly data from one or more INEL meteorological stations are generally available from NOAA beginning in December 1954.

3.1 Operational Releases

For operational conditions, releases from INEL facilities were treated as occurring at a uniform rate over the year. MESODIF calculates an annual average dispersion coefficient that is used to derive the concentration of radioactive material at offsite locations resulting from a uniform annual release rate. Doses are estimated on the basis of these air concentrations. Annual average ground–level air concentrations (expressed in units of Ci/m^3) are calculated from the annual dispersion coefficient (in units of h^2/m^3) by multiplying by the average release rate (in units of Ci/h) and dividing by the number of hours in a year (8760 hours).

The decay of radioactivity during transit from the point of release to the INEL Site boundary was also considered in these calculations. The transport time was based on annual average wind speeds for each year in the direction of each inhabited offsite location where the largest air concentration was predicted. This time of transport accounted for a significant reduction in the annual average ground–level air concentration for short–lived radionuclides.

From 1973 to 1989, with the exception of 1978 when the telemetry system computer was upgraded, adequate meteorological data were available to develop maps of the annual average dispersion coefficients using the MESODIF computer code. During this period, assessments of operational releases were done using the hourly data accumulated over each year. An example of an annual average dispersion coefficient map is shown in Figure 3–1 using the results for 1987. The contours shown are generally elongated along the directions of prevailing winds, reflecting the dispersion of releases to the northeast by daytime winds and southwest by nighttime winds. Because the model uses the actual wind data from a large area, the results also show the effects of recirculation of winds on the Snake River Plain and the influence of mountain valleys to the northwest of the INEL Site.

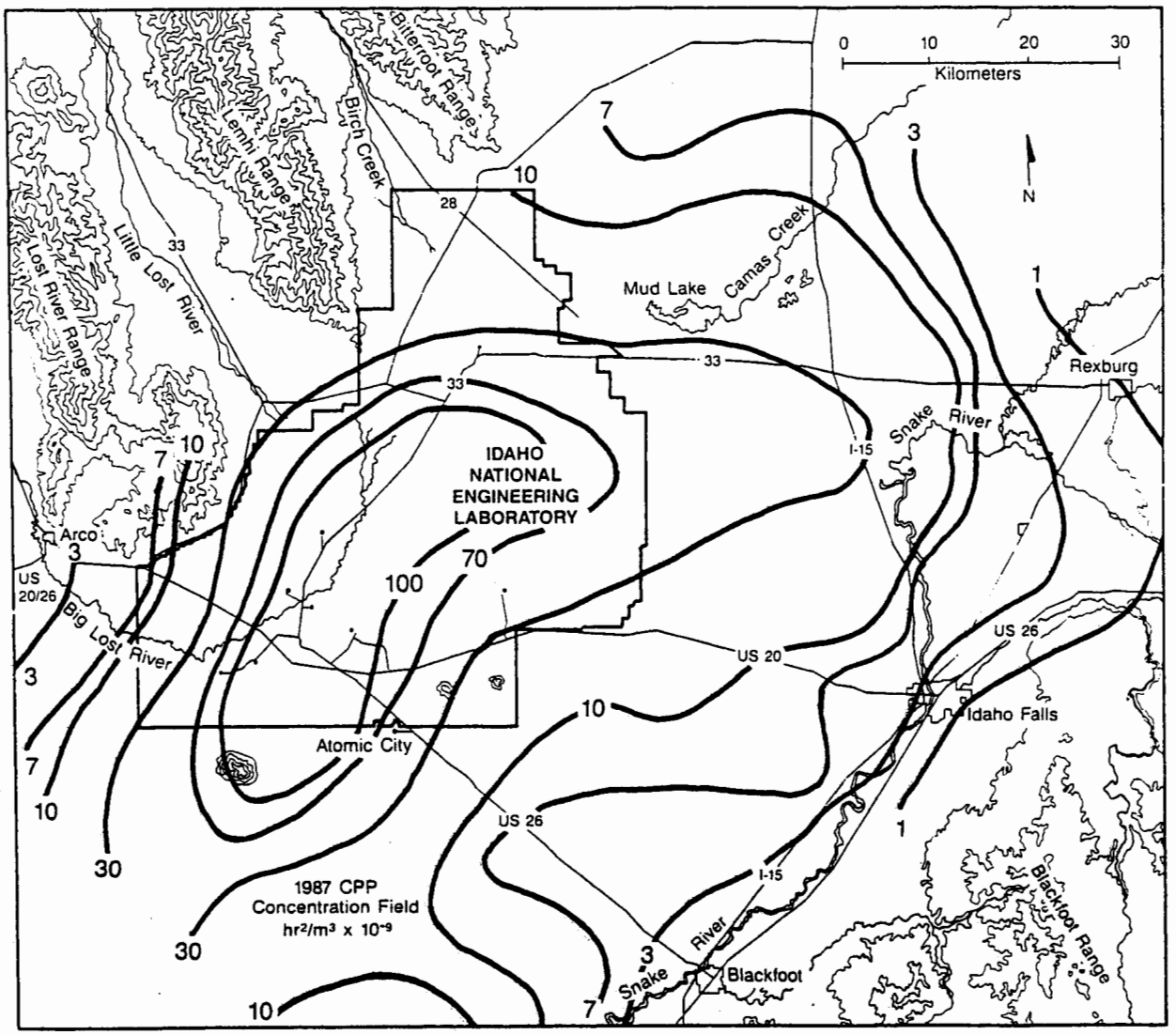


Figure 3–1. Annual average dispersion coefficient contours for 1987.

The inhabited offsite location corresponding to the highest annual average air concentration for 1987 is Atomic City. This result is typical of the results obtained over most years for which annual average dispersion coefficients were calculated. From 1973 to 1989, other locations showed higher values only three times. One location was Terreton in 1973, the second was a point near Hamer in 1977, and the third was at Terreton in 1988.

During 1978 and the period from 1952 to 1972, data for calculating annual average dispersion coefficients are not available from the network of meteorological stations, and individual maps could not be developed using the MESODIF computer code. For this period, the average results based on the best available long–term data set were used. A continuous MESODIF run was made on a 9–year set of hourly wind data for 1974 through 1983, excluding 1978. The results of this calculation are shown in Figure 3–2. The general pattern of the contours is similar to that shown for 1987, with the highest inhabited offsite location being Atomic City. The average value of the annual dispersion coefficient at Atomic City over this 9–year period is 50 x 10^{-9} h^2/m^3. The annual

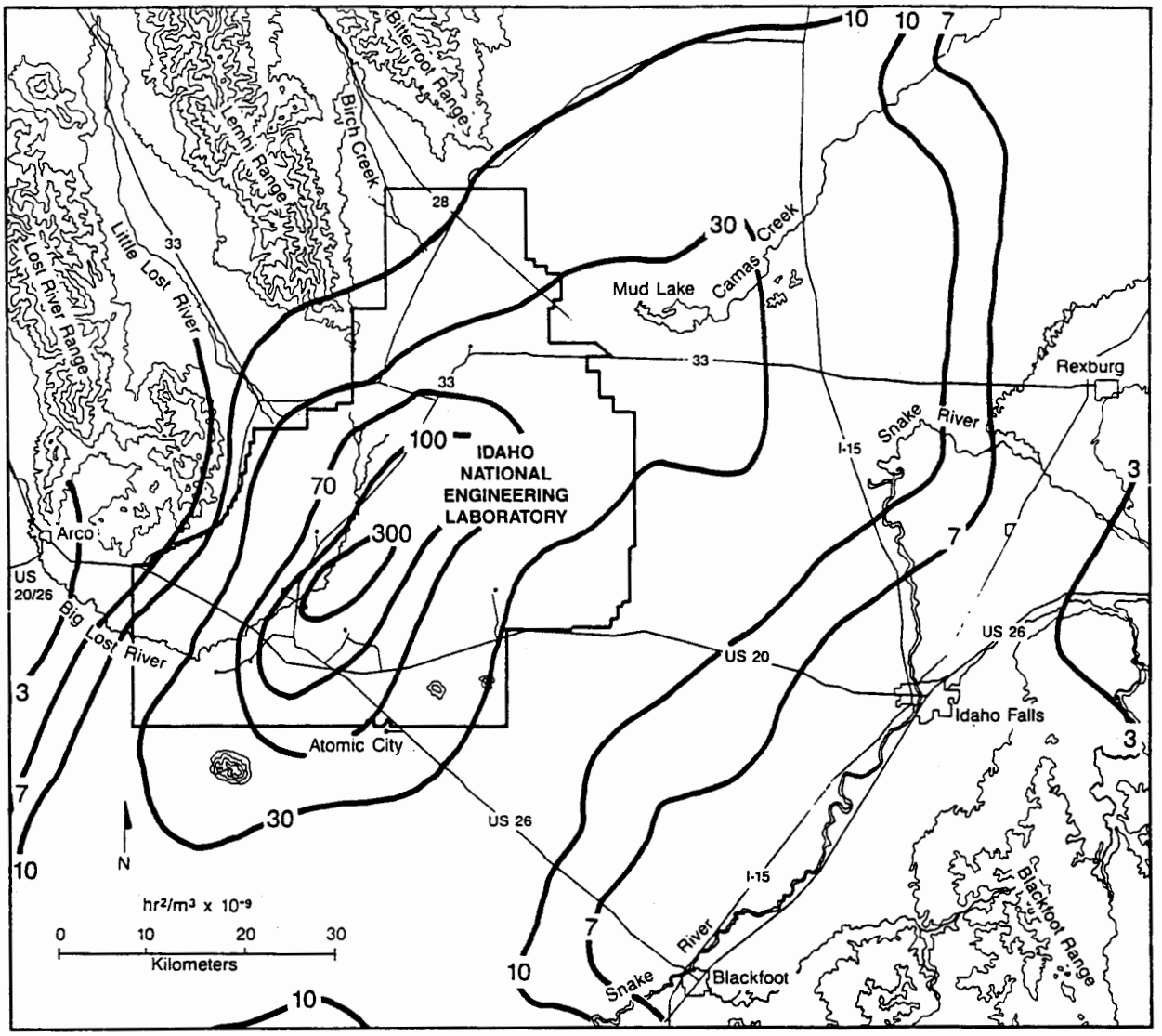


Figure 3–2. Nine–year average dispersion coefficient contours for 1974–1983 (excluding 1978).

dispersion coefficients over this same period (1974–1983) ranged above and below the average by a factor of approximately 2.5, from 20 x 10^{-9} to 120 x 10^{-9} h^2/m^3. This range reflects the variability in the atmospheric dispersion calculation for 1978 and the years before 1973. The 9–year average dispersion factor provides a reasonable estimate that does not appear to consistently over– or underestimate the dispersion of radioactivity released by INEL Site facilities when compared to modeled results for recent years.

3.2 Episodic Releases

It is not appropriate to treat short–term releases as occurring at a uniform rate over the year. The use of annual average dispersion conditions could predict concentrations that are either much smaller or larger than what actually occurred. Further, annual average conditions are unlikely to identify the location of highest offsite concentration for the winds occurring on any particular day. When available, hourly meteorological data for the actual time of release were used for the

episodic releases. If historical meteorological data were incomplete or unavailable in NOAA records, information on meteorological conditions was obtained from written reports.

In either case, the MESODIF computer code was used to calculate the trajectories of individual "puffs" representing the episodic release. Resulting offsite air concentrations were integrated until each puff had dispersed so completely that it no longer made a significant contribution to the total air concentration. Because the atmospheric conditions are continually changing, each puff behaves differently, and each episodic release calculation must be modeled separately for its own time period. Each calculation was continued until the material reached the edge of the calculational grid.

Calculating dispersion coefficients for episodic releases associated with some of the ANP Program tests presented a difficult technical problem. Several of the Initial Engine Tests (IETs) were performed over extended periods of time ranging from a few weeks to several months. Depending on the test protocols, the reactor was operated in a noncontinuous manner at differing times during the day and for varying lengths of time. In some instances, the relative composition of radionuclides in the effluent also changed as the test progressed. Two methods were applied to estimate the dispersion coefficient for the extended tests. If detailed information concerning reactor operations, release rates, and release times was not available, the test was divided into one or more operational periods, and an average dispersion coefficient was calculated over the entire length of each period. This method was used for IETs #4, #6, and #16. When detailed information concerning reactor operations and radioactive releases was available, dispersion factors were calculated for each of the individual runs within a test. A weighted–average dispersion coefficient was then calculated for the test. The weighted–average dispersion coefficient was equal to the sum of products of the dispersion coefficient for each run multiplied by the total radioactivity released during the run (expressed as 10–minute–decayed curies) divided by the total radioactivity released during the test. If the relative composition of radionuclides in the effluent changed during a test, then the test was divided into multiple periods, and weighted–average dispersion coefficients were calculated for each period of operation. Weighted–average dispersion coefficients were calculated for IETs #3, #8, #10, #11, #14, #15, #17 through #21, #23, #25, and #26. IETs #12, #13, #22, and #24 were short duration releases and were modeled as such.

The short–term dispersion coefficient is used to estimate the average air concentration over the time period of the calculation in a manner slightly different from the way the annual average dispersion coefficient is used to estimate the average air concentration over the year. Annual average air concentration depends on the annual release rate from each operating facility. Air concentration from episodic releases depends on the amount of radioactivity released in a limited number of puffs. All short–term dispersion coefficients were normalized for a 1–Ci release even if the release occurred over an extended period. Consequently, the units of the short–term dispersion coefficient are different from those of the annual dispersion coefficient. To obtain an average ground–level air concentration (in units of Ci/m^3), the short–term dispersion coefficient (expressed in units of h/m^3) is multiplied by the amount of the release (in units of Ci) and divided by the time period of the calculation (in units of h), shown as hours modeled in Table 3–1.

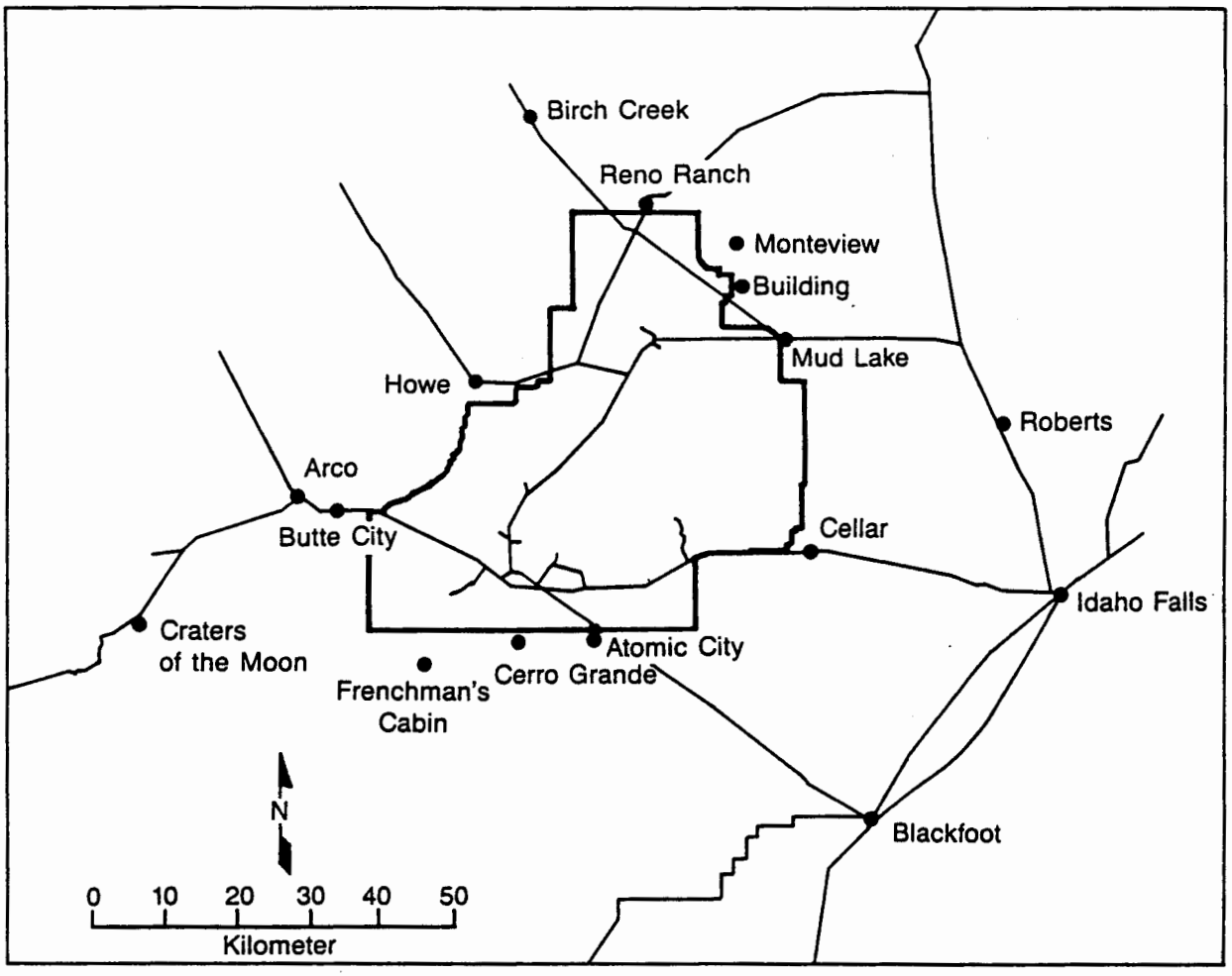
To obtain a dispersion coefficient for a reasonable number of locations around the INEL boundary, 16 points were chosen. Most of these points represent locations where there is reason to believe someone lived in the early days of the INEL Site. There are three locations that were chosen to fill in the spacing along the perimeter that probably do not represent occupied residences. They are (1) Cerro Grande, an abandoned railroad stop; (2) Frenchman's Cabin, an abandoned cabin near Big Southern Butte; and (3) Cellar, a potato cellar southeast of the INEL Site. The locations and titles are shown in Figure 3–3. Table 3–1 presents the locations at

Table 3–1. Dispersion factors for selected episodic releases[a]

Date	Description	Highest Dispersion Factor (h/m^3)	Location	Time to INEL Site Boundary (h)	Hours Modeled
1/3/61	SL–1 Accident	8.00E–11	Atomic City	1.0	16
3/1–3/30/61	IET #26(B)	3.32E–12	Howe	Var[b]	361
9/7–10/14/60	IET #23 (FEET #2)	3.37E–12	Monteview	Var	310.4
5/14–6/10/60	IET #20	1.75E–12	Blackfoot	Var	371.2
3/1–4/30/60	IET #19(B)	3.14E–12	Roberts	Var	553.8
1/6–2/7/60	IET #18	5.47E–12	Cerro Grande	Var	275
10/16/59	1959 ICPP Criticality	2.07E–11	Near Frenchman's Cabin	0.5	2.9
4/24–5/19/59	IET #14	2.12E–12	Frenchman's Cabin	Var	241.3
11/18/58	IET #13 (HTRE No. 3 Excursion)	4.77E–11	Howe	1.4	5.9
10/29–10/30/58	FECF Filter Break	4.83E–11	Frenchman's Cabin	1.4	5.9
9/26/58	FPFRT–9	3.14E–11	Atomic City	15.4	55.9
9/18/58	FPFRT–8	5.83E–10	Mud Lake	14.1	22.9
9/17/58	FPFRT–7	4.34E–11	Mud Lake	21.9	39.9
8/27/58	FPFRT–5	4.63E–11	Mud Lake	5.8	16.9
8/14/58	FPFRT–4	9.75E–11	Mud Lake	3.3	27.9
8/6/58	FPFRT–3	8.33E–11	Cellar	10.0	26.9
8/4/58	FPFRT–2	3.05E–12	Blackfoot	2.2	8.9
3/20–4/14/58	IET #11	1.48E–12	Cellar	Var	260
3/1–3/6/58	IET #10(B)	2.72E–12	Cellar	Var	423.5
12/20/57–2/25/58	IET #10(A)	5.32E–12	Howe	Var	150.7
3/20/57	Fuel Element Burn Test, B	1.58E–10	Birch Creek	2.8	3.9
6/29/56	IET #4(C)	2.50E–12	Monteview	0.3	0.9
5/24–6/26/56	IET #4(B)	1.55E–12	Building	0.7	874
5/1–5/23/56	IET #4(A)	2.24E–12	Building	1.1	545
2/11–2/24/56	IET #3	1.32E–10	Building	0.8	34.6
6/18–7/1/55	NRF S1W Engineering Test	1.40E–11	Building	3.7	349.9

a. Episodic releases producing estimated doses less than 0.1 mrem are not included in this table.

b. Var = varies throughout the test series. Releases for each run of the test series were modeled and decayed according to the transit time to the INEL Site boundary.



9-8199

Figure 3–3. Locations for dispersion calculations for episodic releases.

which the highest integrated air concentrations were calculated for all episodic releases resulting in offsite doses greater than 0.1 mrem. Short–term dispersion coefficients and the modeled time for each calculation are also listed.

Meteorological conditions can result in dramatically different behavior among episodic releases. As a result, the travel time to the INEL Site boundary and the modeled time for the calculation varied among the episodic releases. For short–term releases of 1 hour or less, total time for these calculations was as short as 0.9 hour when winds were strong and persistent and as long as 56 hours when winds were light and variable allowing the puffs to meander during their atmospheric movements.

For example, Figure 3–4 shows the puff trajectory and Figure 3–5 shows the short–term dispersion coefficients for IET #24 (LIME–II), which occurred on October 26, 1960. In this case, the trajectory follows a relatively straight line northeast of the INEL Site. The puff takes only 0.4 hour to reach the INEL Site boundary and provides no significant contribution to the integrated concentration after 0.9 hour. The contours shown in Figure 3–5 indicate the values of the short–term dispersion coefficients for this test. The two contours on this map represent about three orders of magnitude (a factor of 1000) difference in air concentration. The highest integrated offsite concentration at an inhabited location for this example occurred at a point designated "Building."

In contrast, Figures 3–6 and 3–7 show the puff trajectory and short–term dispersion coefficients for the Fission Product Field Release Test No. 8 (FPFRT–8) on September 19, 1958. This puff also followed a straight path but traveled much more slowly than in the previous example, taking 14.1 hours to reach the INEL Site boundary and requiring modeled computations over a total of 22.9 hours. The release traveled to the northeast, and the highest offsite concentration occurred at Mud Lake. The longer travel time allowed for greater dispersion.

Figures 3–8 and 3–9 illustrate that puff trajectories often meander quite dramatically and cannot always be approximated by a straight line based on wind conditions at the point of release. Conditions on August 6, 1958, following the FPFRT–3, resulted in an initial trajectory toward the northeast that subsequently wandered over much of the southeast portion of the INEL Site. The test cloud then traveled in a southwesterly direction down the Snake River Plain and finally reversed direction to the northeast. Travel time to the INEL Site boundary was 10 hours, and the total time of integration was 27 hours. The highest offsite concentrations occurred at a point near Highway US 20, and doses were calculated at a potato cellar near that location.

Other examples of important releases (IET #3; Fuel Element Burn Test, B; and the SL–1 Accident) are shown in Figures 3–10 through 3–14. The dispersion map for the SL–1 Accident differs from the other maps because it is based on extensive environmental sampling at the time of the accident and not a modeled dispersion estimate. This is the only episodic release where the Task Group believed the environmental data were sufficiently complete to warrant their use, rather than a dispersion factor calculated from meteorological data.

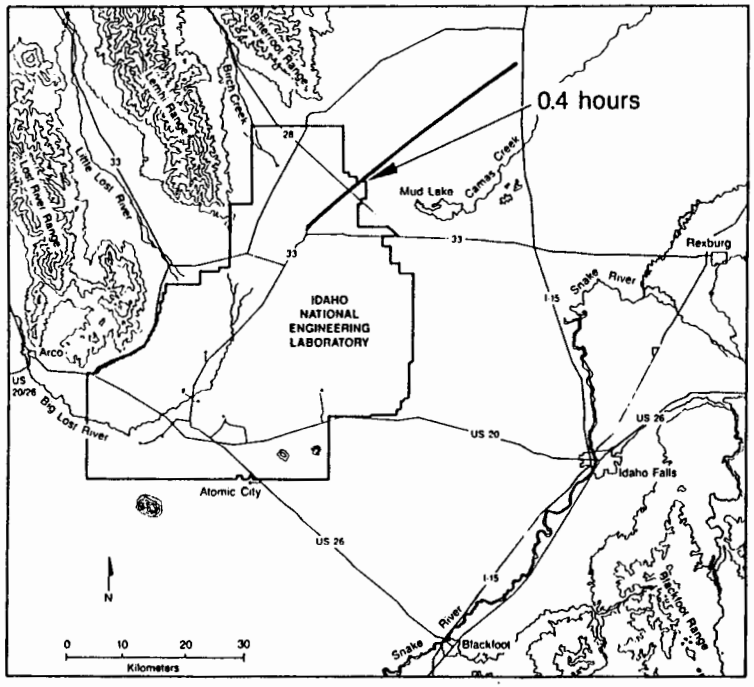


Figure 3–4. Release trajectory for IET #24 (LIME–II).

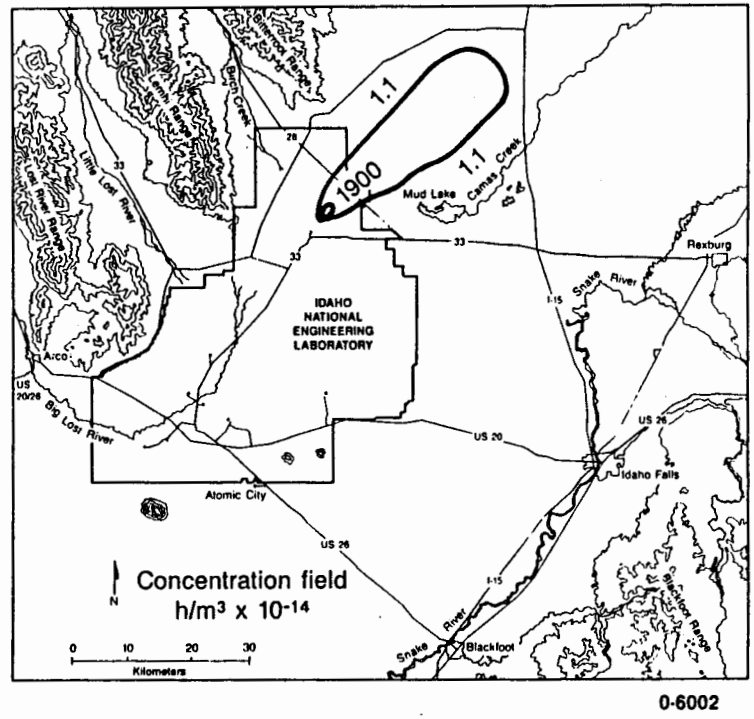
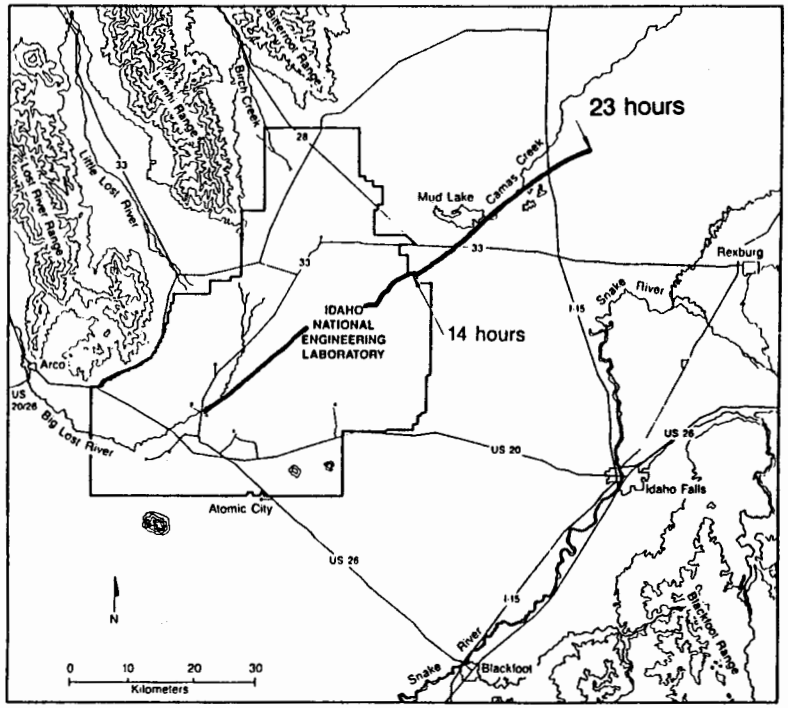
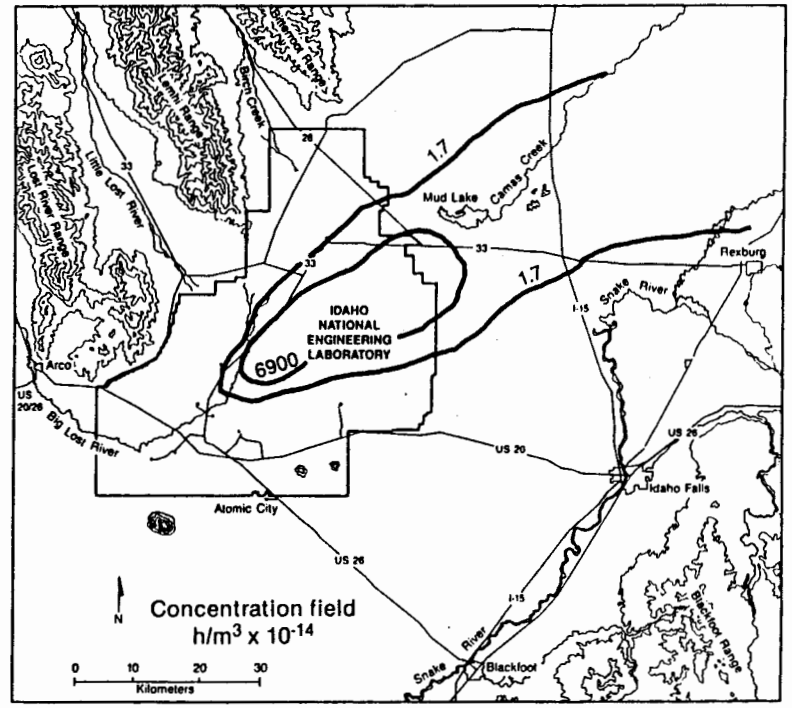


Figure 3–5. Dispersion coefficient contours for IET #24 (LIME–II).



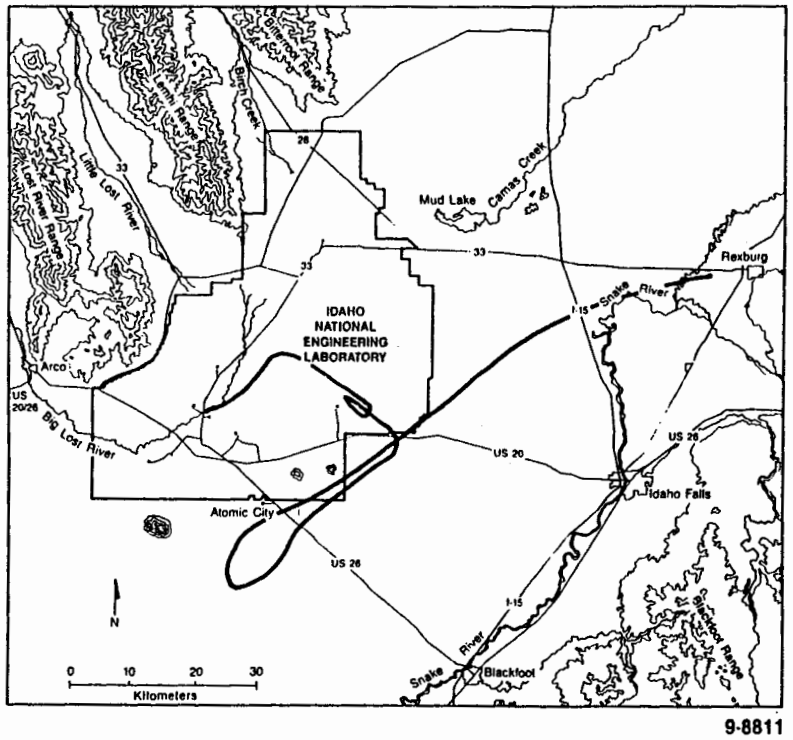
9-8320

Figure 3–6. Release trajectory for FPFRT–8.



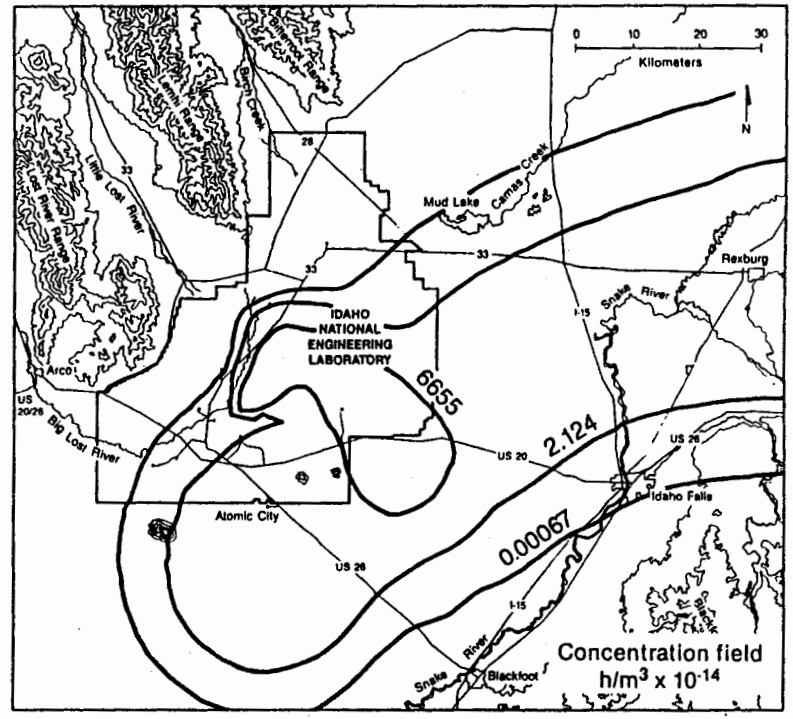
9-8304

Figure 3–7. Dispersion coefficient contours for FPFRT–8.



9-8811

Figure 3–8. Release trajectory for FPFRT–3.



9-8810

Figure 3–9. Dispersion coefficient contours for FPFRT–3.

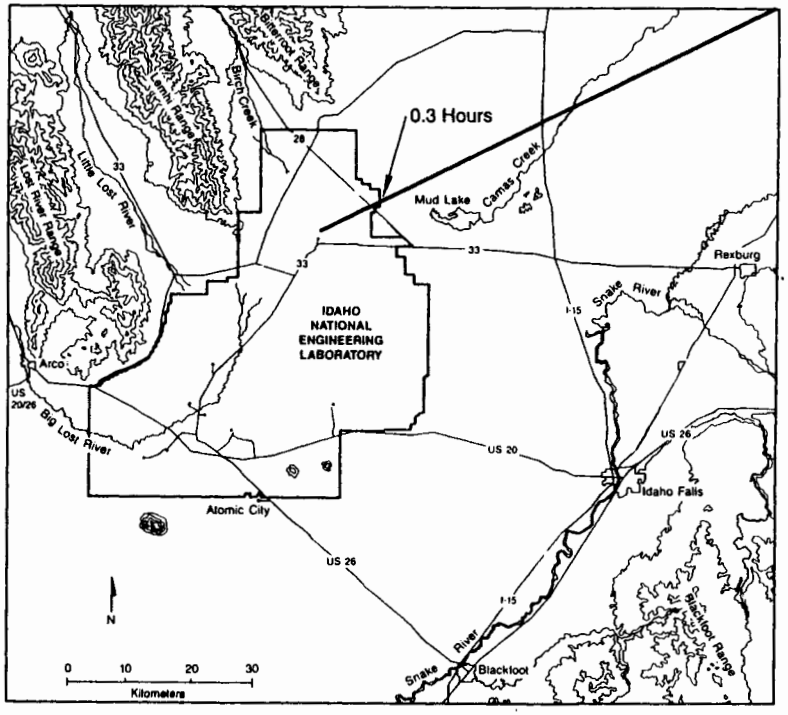


Figure 3–10. Release trajectory for IET #3.

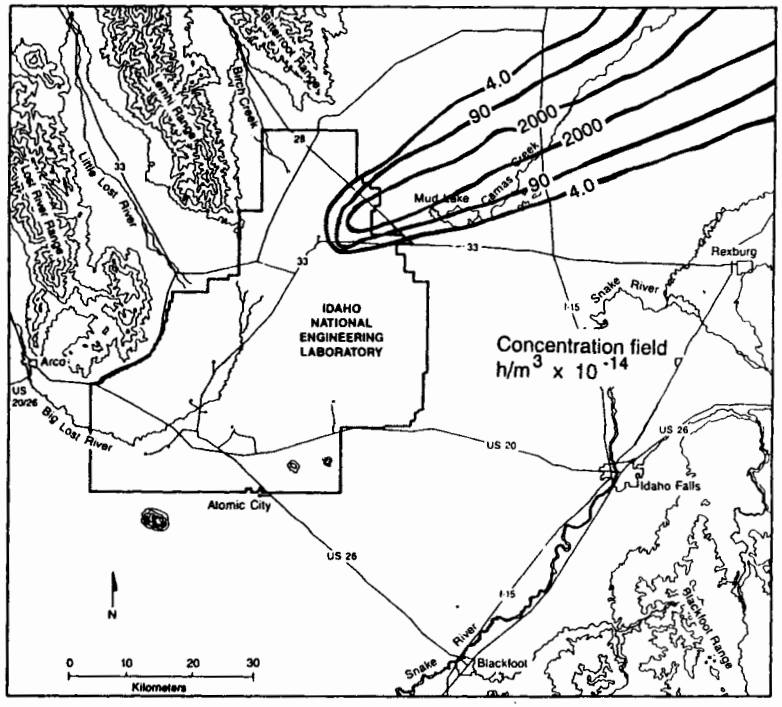


Figure 3–11. Dispersion coefficient contours for IET #3.

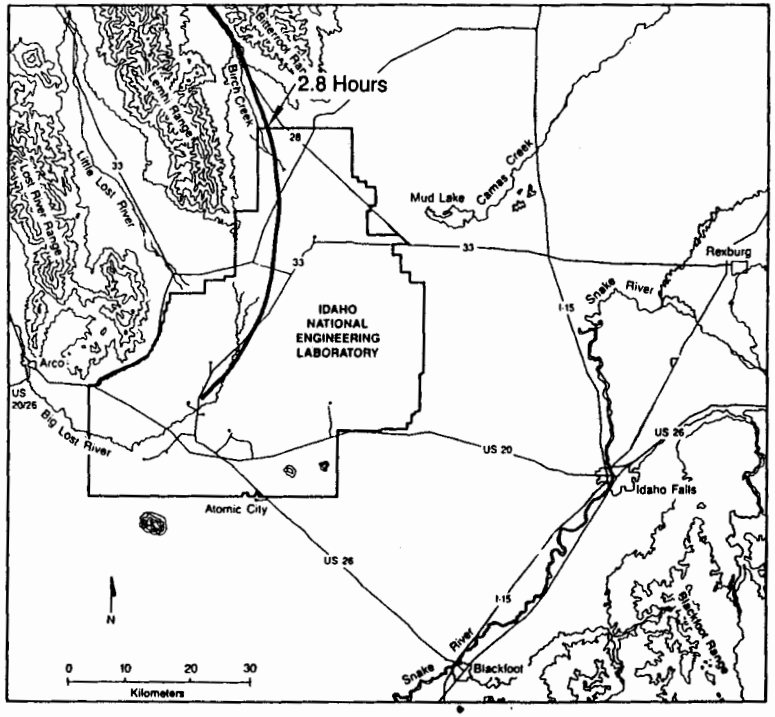


Figure 3–12. Release trajectory for Fuel Element Burn Test, B.

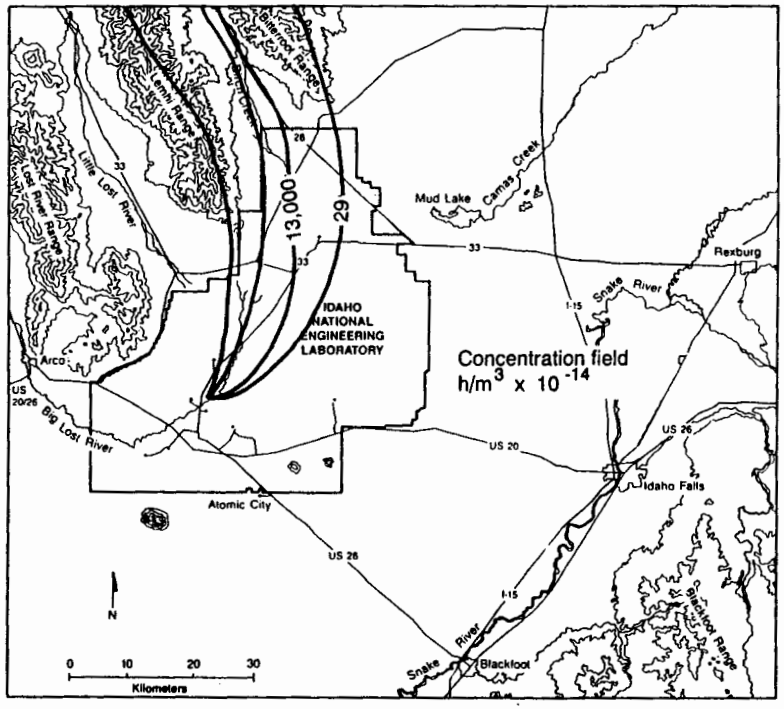


Figure 3–13. Dispersion coefficient contours for Fuel Element Burn Test, B.

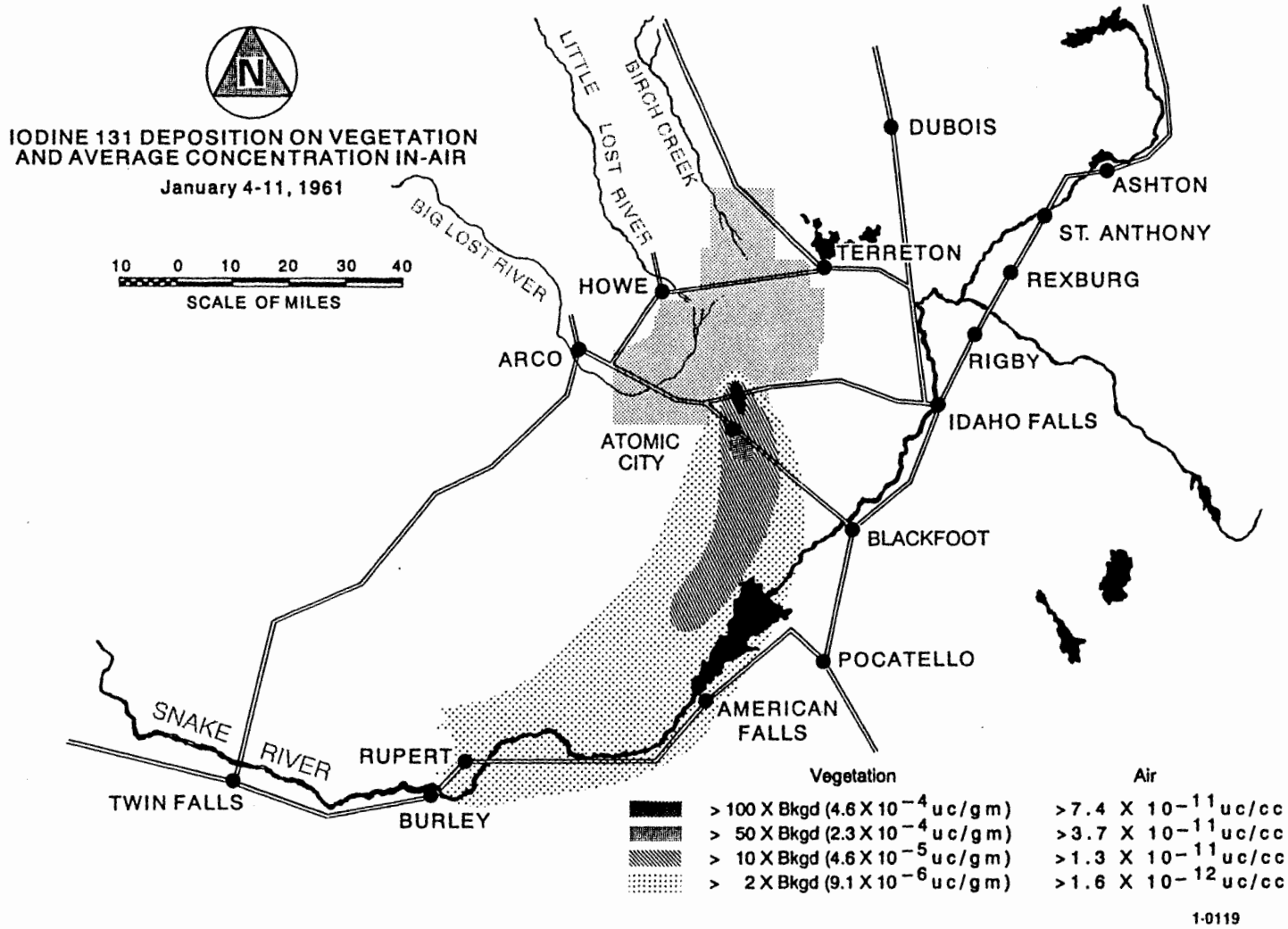


Figure 3–14. Dispersion coefficient contours for the SL–1 Accident.

4. DOSE EVALUATION

4.1 Dosimetric Considerations

Doses were calculated for each year of operation and for most episodic releases. The hypothetical individual (adult, child, or infant), for whom the dose calculation was performed, was assumed to reside at a location near the INEL Site boundary for the duration of the release. For annual operational releases, the nearest inhabited offsite location corresponding to the highest annual average concentration of radioactivity in air was chosen. For episodic releases, the nearest inhabited offsite location corresponding to the highest time–integrated air concentrations of radionuclides was used (see Table 3–1).

The hypothetical individual receiving the largest radiation dose for each year was determined by evaluating the combined radiation dose for the operational and episodic releases in that year. For most years of INEL operation, episodic releases did not contribute significantly to offsite dose. Therefore, the inhabited location of maximum annual average air concentration from operational releases and the location of the individual receiving the largest dose are the same. For the years when the episodic releases did contribute significantly to the combined radiation dose from operational and episodic releases, the individual receiving the largest dose was assumed to inhabit the offsite location where the largest sum of the operational and episodic doses occurred. The doses from operational releases were assumed to be directly proportional to the annual dispersion coefficient.

Three age groups were treated in this evaluation: (1) adults, (2) children (10–year old), and (3) infants (0–12 months old). These age groups were chosen because they demonstrate the largest differences in dose to the specific organs from radionuclides of concern at the INEL Site. Differences in dose estimates among age groups arise primarily from two considerations: (1) dose conversion factors for inhaled and ingested radioactivity vary with age, and (2) diets differ with age.

Radiological dose assessment calculations were made for four modes of exposure for the adult, child, and infant: (1) external exposure from immersion in contaminated air, (2) external exposure from radionuclides deposited on ground surfaces, (3) internal exposure from inhalation of contaminated air, and (4) internal exposure from ingestion of contaminated agricultural products. Dose conversion factors for the adult age group for this analysis were taken from compilations prepared for use by DOE in the calculation of dose to the public (DOE, 1988a; DOE, 1988b). These dose conversion factors are based on the most recent recommendations of the International Commission on Radiological Protection (ICRP, 1977; ICRP 1976–1983; ICRP 1986). Age–specific dose conversion factors for ingestion and inhalation of the radionuclides that contribute most of the dose from INEL Site releases were taken from ICRP (1989). The addition of the dose from daughter radionuclides was included with the dose from the parent radionuclide.

Organ doses are reported as the 50–year committed dose equivalent. This means that if a radionuclide is retained in the body for more than 1 year, as in the case of strontium–90, the dose is accrued during a period of 50 years. Total doses are reported as effective dose equivalent (EDE), which is the sum of the dose from external sources and the committed effective dose equivalent from internally deposited radionuclides. This committed effective dose equivalent is the sum of the committed dose equivalents to the organs or tissues of the body, each multiplied by a corresponding weighting factor. The weighting factors used each reflect the relative carcinogenic or genetic risk resulting from exposure of the individual organ or tissue to radioactivity. The EDE is proportional to the risk of detrimental health effects occurring from exposure to radioactive material and provides a common basis for comparing the relative importance of different episodic and operational releases.

4.2 Results of Dose Evaluation

The operational release dose results are summarized in Table 4–1. The episodic release dose results are summarized in Table 4–2 for those releases resulting in an EDE greater than 0.1 mrem. The operational and episodic doses are combined for each year. The combination of doses from episodic and operational releases requires the accounting for each source at all locations, not just the location of the highest air concentration. The results of these summations are shown in Table 4–3. Table 4–3 includes the estimates of doses to the hypothetical individual receiving the largest dose from operational and episodic releases for all years of INEL operation from 1952 to 1989. Values are presented for both the EDE and for the organ receiving the largest dose. Doses are shown for three age groups: adult, child, and infant. The significance of the estimated doses is discussed in Section 4.3. Appendix C to this report contains a complete list of all dose estimates for each year of operational release and each of the episodic releases analyzed. The parameters used in making the dose calculations are listed in Appendix C along with a sample calculation.

The results of the dose calculations are plotted for operational releases for an adult in Figure 4–1. The EDE is plotted as a solid line, and the maximum organ dose is plotted as a dashed line. The estimated EDEs to an adult from operational releases range over three orders of magnitude from 11 mrem in 1957 to 0.006 mrem in 1989. The decreasing trend in EDE since 1957 is due to the lower airborne releases of radioactivity as a result of changes in facility missions, completion of reactor testing programs involving the MTR, improvements in process design, and improvements in emission control systems. The highest EDEs occurred from 1957 through 1959 and were primarily due to the processing of reactor fuels for RaLa at ICPP (see Section A.2.2.4). In 1957, 64% of the EDE was due to iodine–131, 17% was due to cesium–137, and 11% was due to strontium–90. Improved controls reduced the emissions of radioiodines from the RaLa project. By 1959, the iodine–131 contribution to the EDE declined to 19%, while cesium–137 was 38% of the total. From 1960 to 1962, the effluents from the Test Reactor Area (TRA), particularly the MTR, make a substantial contribution to the estimated EDE. Argon–41 accounted for 32% of the EDE in 1962. Krypton–88, strontium–90, iodine–131, and cesium–137 were also important contributors to the EDE during 1962. Cesium–137 was the highest dose contributor to the EDE in 11 of the 21 years from 1954 to 1974. Cesium–137 accounted for a large fraction of the peak EDEs in 1963, 1965, and 1971. Ruthenium–106 was a large contributor in 1964 and 1966. The peak in 1968 was largely due to strontium–90 and cesium–137.

Some variations occur in the calculated doses for operational releases (Figure 4–1) from year to year because of intermittent facility operations or campaigns. For example, during 1989, no fuel was processed in the Fluorinel and Fuel Storage (FAST) facility at the ICPP, and the New Waste Calcining Facility (NWCF) did not operate during most of the year. The Advanced Test Reactor (ATR) operated only about 50% of the time during 1989 because of a console upgrade. During 1987 and 1988, the EDE was higher than in the years before and after. This was caused by releases of antimony–125 during fuel processing at the FAST facility at the ICPP (see Section E.5.1) in addition to the releases of iodine–129. The FAST facility began processing fuel in the last quarter of 1986, and the campaign continued through the first half of 1988. Important processes at the ICPP were not in operation for most of 1985 and 1986. The ATR reactor only operated 60% of 1986; therefore, the resulting doses for 1985 and 1986 are lower.

The maximum organ dose equivalent to an adult from operational releases (see Figure 4–1) follows the same trend as the EDE, increasing from 1952 through 1957 and then generally decreasing through 1989. The largest organ dose equivalent ranges from 230 mrem in 1957 to 0.04 mrem in 1989. In the years between 1952 and 1956 and 1963 through 1975, the dose equivalent to the skin is higher than the dose to the

Table 4–1. Dose estimates for a hypothetical offsite individual from operational releases at the INEL[a]

Year	Effective Dose Equivalent (mrem)			Radiation Protection Standard[b] (mrem/yr)	Maximum Organ Dose (mrem)						Radiation Protection Standard (mrem/yr)
	Adult	Child	Infant		Adult	Organ	Child	Organ	Infant	Organ	
1989	0.006	0.006	0.006	25	0.04	Skin	0.04	Skin	0.04	Skin	75
1988	0.15	0.17	0.14	25	2	Thyroid	3	Thyroid	1.8	Thyroid	75
1987	0.11	0.12	0.10	25	0.9	Thyroid	1.2	Thyroid	0.9	Thyroid	75
1986	0.03	0.03	0.03	25	0.11	Thyroid	0.13	Thyroid	0.10	Thyroid	75
1985	0.02	0.03	0.02	25	0.18	Skin	0.18	Skin	0.18	Skin	75
1984	0.03	0.03	0.02	500	0.5	Thyroid	0.7	Thyroid	0.5	Thyroid	1500
1983	0.04	0.05	0.04	500	0.7	Thyroid	0.9	Thyroid	0.6	Thyroid	1500
1982	0.03	0.04	0.03	500	0.4	Thyroid	0.5	Thyroid	0.4	Thyroid	1500
1981	0.04	0.04	0.04	500	0.3	Thyroid	0.4	Thyroid	0.3	Thyroid	1500
1980	0.08	0.08	0.07	500	0.6	Thyroid	0.9	Thyroid	0.6	Thyroid	1500
1979	0.07	0.07	0.07	500	0.5	Thyroid	0.6	Thyroid	0.4	Thyroid	1500
1978	0.08	0.09	0.07	500	0.8	Thyroid	1.0	Thyroid	0.7	Thyroid	1500
1977	0.13	0.13	0.11	500	1.0	Thyroid	1.3	Thyroid	0.9	Thyroid	1500
1976	0.3	0.3	0.3	500	1.3	Thyroid	1.6	Thyroid	1.2	Thyroid	1500
1975	0.17	0.17	0.16	500	0.6	Skin	0.7	Thyroid	0.7	Skin	1500
1974	0.3	0.2	0.2	500	1.7	Skin	1.7	Skin	1.7	Skin	1500
1973	0.3	0.3	0.3	500	1.8	Skin	1.8	Skin	1.8	Skin	1500
1972	0.5	0.5	0.5	500	4	Skin	4	Skin	4	Skin	1500
1971	1.8	1.8	1.9	500	17	Skin	17	Skin	17	Skin	1500
1970	0.4	0.4	0.4	500	4	Skin	4	Skin	4	Skin	1500
1969	0.7	0.6	0.7	500	5	Skin	5	Skin	5	Skin	1500
1968	1.4	1.4	1.4	500	16	Skin	16	Skin	16	Skin	1500
1967	0.8	0.8	0.8	500	5	Skin	5	Skin	5	Skin	1500
1966	2	2	2	500	20	Skin	21	Skin	21	Skin	1500
1965	4	3	4	500	35	Skin	35	Skin	35	Skin	1500
1964	3	4	4	500	31	Skin	31	Skin	32	Skin	1500
1963	5	4	5	500	37	Skin	37	Skin	37	Skin	1500
1962	1.4	1.4	1.9	500	8	Thyroid	12	Thyroid	27	Thyroid	1500
1961	1.3	1.4	1.8	500	8	Thyroid	12	Thyroid	28	Thyroid	1500
1960	1.0	1.1	1.4	500	6	Thyroid	9	Thyroid	21	Thyroid	1500
1959	6	6	9	500	45	Skin	63	Thyroid	140	Thyroid	1500
1958	11	13	25	500	170	Thyroid	280	Thyroid	650	Thyroid	1500
1957	11	15	30	500	230	Thyroid	370	Thyroid	880	Thyroid	1500
1956	3	3	4	1500	34	Skin	34	Skin	34	Skin	2500
1955	3	3	3	1500	32	Skin	32	Skin	32	Skin	2500
1954	2	2	2	1500	23	Skin	23	Skin	23	Skin	2500
1953	1.6	1.6	1.6	1500	12	Skin	12	Skin	12	Skin	2500
1952	0.4	0.4	0.4	1500	2	Skin	2	Skin	2	Skin	2500

a. Doses are calculated for location of highest annual average atmospheric concentration. This corresponds to Atomic City except for 1973 (near Terreton), 1977 (8 mi W of Hamer), and 1988 (near Terreton).

b. The radiation protection standard shown is that for the whole body and is not strictly comparable to the calculated EDE. The EDE provides a common basis for comparing the importance of various releases.

Table 4–2. Dose estimates for a hypothetical offsite individual from selected episodic releases at the INEL[a,b]

Dates	Episodes	Effective Dose Equivalent (mrem)			Maximum Organ Dose (mrem)					
		Adult	Child	Infant	Adult	Organ	Child	Organ	Infant	Organ
1/3/61	SL–1 Accident	3	4	7	39	Thyroid	71	Thyroid	180	Thyroid
3/1–30/61	IET #26(B)	0.04	0.06	0.15	0.9	Thyroid	1.6	Thyroid	5	Thyroid
9/7–10/14/60	IET #23 (FEET #2)	0.03	0.05	0.13	0.8	Thyroid	1.5	Thyroid	4	Thyroid
5/14–6/10/60	IET #20	0.04	0.06	0.13	0.8	Thyroid	1.5	Thyroid	4	Thyroid
3/1–4/30/60	IET #19(B)	0.08	0.12	0.3	1.9	Thyroid	3	Thyroid	9	Thyroid
1/6–2/7/60	IET #18	0.10	0.11	0.18	0.9	Thyroid	1.5	Thyroid	3	Thyroid
10/16/59	1959 ICPP Criticality	1.1	1.2	1.5	6	Thyroid	9	Thyroid	22	Thyroid
4/24–5/19/59	IET #14	0.08	0.12	0.3	2	Thyroid	4	Thyroid	10	Thyroid
11/18/58	IET #13 (HTRE No. 3 Excursion)	0.12	0.12	0.16	0.8	Thyroid	1.2	Thyroid	3	Thyroid
10/29–30/58	FECF Filter Break	0.11	0.12	0.12	1.4	Skin	1.4	Skin	1.4	Skin
9/26/58	FPFRT–9	1.6	1.5	1.5	12	Skin	12	Skin	12	Skin
9/18/58	FPFRT–8	14	15	17	66	Skin	72	Thyroid	190	Thyroid
9/17/58	FPFRT–7	0.9	1.0	1.3	4	Skin	6	Thyroid	15	Thyroid
8/27/58	FPFRT–5	1.7	2	4	18	Thyroid	34	Thyroid	97	Thyroid
8/14/58	FPFRT–4	4	4	4	32	Skin	32	Skin	32	Skin
8/6/58	FPFRT–3	4	4	4	29	Skin	29	Skin	29	Skin
8/4/58	FPFRT–2	0.13	0.12	0.13	1.0	Skin	1.0	Skin	1.0	Skin
3/20–4/14/58	IET #11	0.03	0.05	0.14	0.9	Thyroid	1.6	Thyroid	4	Thyroid
3/1–6/58	IET #10(B)	0.4	0.5	0.8	3	Thyroid	6	Thyroid	16	Thyroid
12/20/57–2/25/58	IET #10(A)	0.7	0.7	0.8	2	Thyroid	3	Thyroid	6	Thyroid
3/20/57	Fuel Element Burn Test, B	5	6	6	60	Skin	60	Skin	60	Skin
6/29/56	IET #4(C)	4	6	13	70	Thyroid	140	Thyroid	370	Thyroid
5/24–6/26/56	IET #4(B)	1.6	2	4	28	Thyroid	44	Thyroid	110	Thyroid
5/1–23/56	IET #4(A)	0.07	0.09	0.18	1.3	Thyroid	2	Thyroid	5	Thyroid
2/11–24/56	IET #3	29	34	54	320	Thyroid	510	Thyroid	1200	Thyroid
6/18–7/1/55	NRF S1W Engineering Test	0.11	0.2	0.5	3	Thyroid	6	Thyroid	18	Thyroid

a. Episodic releases producing estimated doses less than 0.1 mrem are not included in this table.

b. Doses are calculated for location of highest integrated atmospheric concentration. Locations for each episode can be found in Table 2–1.

Table 4–3. Estimates of the total dose for a hypothetical offsite individual from operational and episodic releases at the INEL (1989 through 1952)

Year	Location	Effective Dose Equivalent (mrem)			Radiation Protection Standard[a] (mrem/yr)	Maximum Organ Dose (mrem)						Radiation Protection Standard (mrem/yr)
		Adult	Child	Infant		Adult	Organ	Child	Organ	Infant	Organ	
1989	Atomic City	0.006	0.006	0.006	25	0.04	Skin	0.04	Skin	0.04	Skin	75
1988	Terreton	0.15	0.17	0.14	25	2	Thyroid	3	Thyroid	1.8	Thyroid	75
1987	Atomic City	0.11	0.12	0.10	25	0.9	Thyroid	1.2	Thyroid	0.9	Thyroid	75
1986	Atomic City	0.03	0.03	0.03	25	0.11	Thyroid	0.13	Thyroid	0.10	Thyroid	75
1985	Atomic City	0.02	0.03	0.02	25	0.18	Skin	0.18	Skin	0.18	Skin	75
1984	Atomic City	0.03	0.03	0.02	500	0.5	Thyroid	0.7	Thyroid	0.5	Thyroid	1500
1983	Atomic City	0.04	0.05	0.04	500	0.7	Thyroid	0.9	Thyroid	0.6	Thyroid	1500
1982	Atomic City	0.03	0.04	0.03	500	0.4	Thyroid	0.5	Thyroid	0.4	Thyroid	1500
1981	Atomic City	0.04	0.04	0.04	500	0.3	Thyroid	0.4	Thyroid	0.3	Thyroid	1500
1980	Atomic City	0.08	0.08	0.07	500	0.6	Thyroid	0.9	Thyroid	0.6	Thyroid	1500
1979	Atomic City	0.07	0.07	0.07	500	0.5	Thyroid	0.6	Thyroid	0.4	Thyroid	1500
1978	Atomic City	0.08	0.09	0.07	500	0.8	Thyroid	1.0	Thyroid	0.7	Thyroid	1500
1977	Hamer	0.13	0.13	0.11	500	1.0	Thyroid	1.3	Thyroid	0.9	Thyroid	1500
1976	Atomic City	0.3	0.3	0.3	500	1.3	Thyroid	1.6	Thyroid	1.2	Thyroid	1500
1975	Atomic City	0.17	0.17	0.16	500	0.6	Skin	0.7	Thyroid	0.7	Skin	1500
1974	Atomic City	0.3	0.2	0.2	500	1.7	Skin	1.7	Skin	1.7	Skin	1500
1973	Terreton	0.3	0.3	0.3	500	1.8	Skin	1.8	Skin	1.8	Skin	1500
1972	Atomic City	0.5	0.5	0.5	500	4	Skin	4	Skin	4	Skin	1500
1971	Atomic City	1.8	1.8	1.9	500	17	Skin	17	Skin	17	Skin	1500
1970	Atomic City	0.4	0.4	0.4	500	4	Skin	4	Skin	4	Skin	1500
1969	Atomic City	0.7	0.6	0.7	500	5	Skin	5	Skin	5	Skin	1500
1968	Atomic City	1.4	1.4	1.4	500	16	Skin	16	Skin	16	Skin	1500
1967	Atomic City	0.8	0.8	0.8	500	5	Skin	5	Skin	5	Skin	1500
1966	Atomic City	2	2	2	500	20	Skin	21	Skin	21	Skin	1500
1965	Frenchman's Cabin	4	3	4	500	35	Skin	35	Skin	35	Skin	1500
1964	Atomic City	3	4	4	500	31	Skin	31	Skin	32	Skin	1500
1963	Atomic City	5	4	5	500	37	Skin	37	Skin	37	Skin	1500
1962	Atomic City	1.4	1.4	1.9	500	8	Thyroid	12	Thyroid	27	Thyroid	1500
1961[b]	Atomic City	4	6	9	500	48	Thyroid	84	Thyroid	210	Thyroid	1500
1960	Atomic City	1.2	1.3	1.8	500	9	Thyroid	14	Thyroid	320	Thyroid	3000
1959	Frenchman's Cabin	7	8	11	500	48	Thyroid	77	Thyroid	180	Thyroid	3000
1958	Mud Lake	29	32	45	500	190	Thyroid	320	Thyroid	780	Thyroid	3000
1957	Atomic City	11	15	30	500	230	Thyroid	370	Thyroid	880	Thyroid	1500
1956	Building	33	38	61	1500	350	Thyroid	560	Thyroid	1350	Thyroid	1500
1955	Frenchman's Cabin	4	4	4	1500	32	Skin	32	Skin	32	Skin	2500
1954	Frenchman's Cabin	2	2	2	1500	23	Skin	23	Skin	23	Skin	2500
1953	Atomic City	1.6	1.6	1.6	1500	12	Skin	12	Skin	12	Skin	2500
1952	Atomic City	0.4	0.4	0.4	1500	2	Skin	2	Skin	2	Skin	2500

a. The radiation protection standard shown is that for the whole body and is not strictly comparable to the calculated EDE. The EDE provides a common basis for comparing the importance of various releases.

b. Before 1961, the estimates of total dose are all included in the operational release estimates (see Table 4–1).

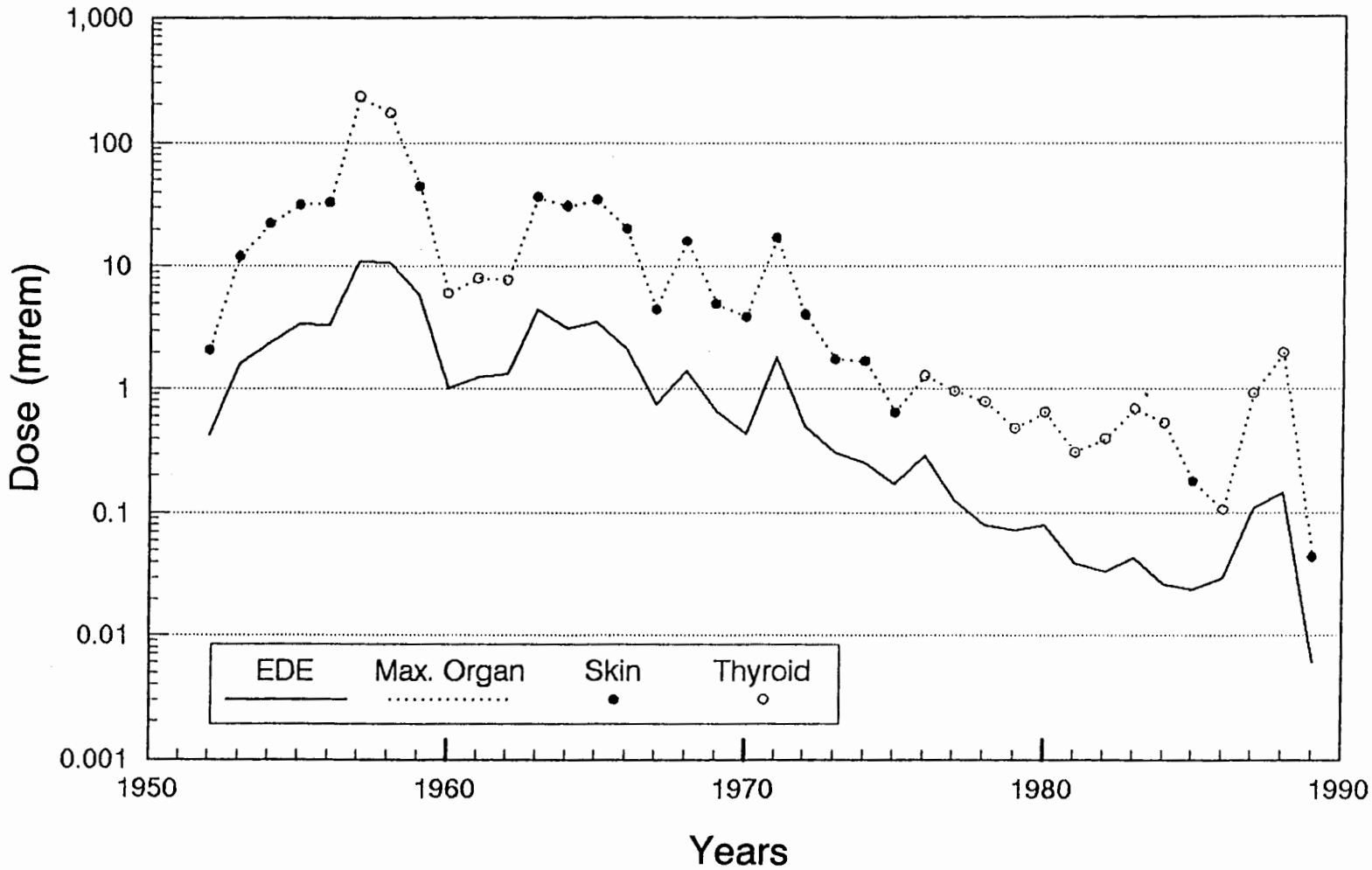


Figure 4–1. Maximally exposed adult operational dose estimates.

thyroid and other organs. For these years, external exposure pathways from air immersion and ground deposition usually dominate; the major contributors to the skin dose were strontium–90, cesium–137, cerium–144, and (in 1964, 1966, and 1971) ruthenium–106. Noble gases, such as argon–41, krypton–88, and krypton–85, are also contributors to the skin dose during these years. From 1957 through 1962 (except in 1959), the largest organ dose was received by the thyroid, primarily as a result of iodine–131 releases from RaLa. For most of the years from 1976 to 1989, the thyroid also has the largest dose equivalent. However, the primary contributor to the thyroid dose is long–lived iodine–129 from fuel processing and waste calcination, rather than the iodine–131 characteristic of the 1950s. In August 1975, new emission control filters, referred to as the Atmospheric Protection System, were installed on the main stack at the ICPP. This system reduced particulate emissions, including strontium–90, cesium–137, cerium–144, and ruthenium–106, resulting in the shift from the skin to the thyroid as the organ with the largest annual dose.

Figure 4–2 shows the EDE and maximum organ doses for the infant from operational releases. As with the adult, the EDE and the maximum organ doses tend to increase from 1952 through 1957 and then generally decrease from 1957 to 1989. The infant EDE ranges from 30 mrem in 1957 to 0.006 mrem in 1989, while the largest organ dose equivalent ranges from 880 mrem in 1957 to 0.04 mrem in 1989. In general, the same radionuclides that produce the majority of the dose to an adult also produce the majority of the dose to the infant. For those years in which iodine–131 releases from RaLa were dominant (1957 through 1962), the thyroid doses are higher for the infant than for the adult because of the smaller thyroid and a larger fractional iodine uptake than adults, which results in a larger ingestion dose conversion factor for iodine–131. The thyroid doses calculated for an infant assume that an infant is fed milk from cows grazing near the INEL Site boundary. No reductions in iodine–131 intake are assumed for infants consuming milk from other sources such as breast milk, formula, or milk collected regionally by commercial dairies.

The dose estimates for a child were not plotted because they are similar to those received by an adult. One variation in the doses between the adult and the child results from age–dependent differences in the ingestion dose conversion factors for iodine–129. The dose equivalent to the thyroid of a child from ingestion of iodine–129 exceeds the dose equivalent calculated for the adult thyroid. The dose to a child's thyroid is about 35% higher than the values shown in Figure 4–1 for the adult thyroid from 1976 through 1988.

Figure 4–3 presents an overlay of the information contained in Figures 4–1 and 4–2. Except from 1957 to 1962, the EDEs and largest organ doses for the infant and adult are quite similar. The difference from 1957 through 1962 is primarily due to the iodine–131 releases during RaLa resulting in higher thyroid dose equivalent and EDE for an infant compared to an adult.

The episodic release dose results were shown in Table 4–2. The highest EDEs for episodic releases were for IET #3; Fuel Element Burn Test, B; FPFRT–8; IET #4(C); and the Stationary Low–Power Reactor No. 1 (SL–1) Accident. Most of the other tests with an EDE above 0.1 mrem are in the IET and FPFRT series. All of the episodic releases above 0.1 mrem EDE were experiments performed under controlled conditions except the SL–1 Accident, the 1959 ICPP Criticality, the IET #13 (HTRE No. 3 Excursion), and the FECF Filter Break. The EDEs for the episodic releases range from 54 mrem (infant) for the IET #3 to less than 0.001 mrem for some of the tests not shown here (see Appendix C). The maximum organ doses range from 1230 mrem to less than 0.001 mrem (see Appendix C). Most of the maximum organ doses are to the thyroid; in those cases, the infant is the age group with the highest organ dose. The skin is the maximum organ for eight episodic releases shown. The most important radionuclide for IET #3 and the SL–1 Accident was iodine–131. For the FPFRT–8

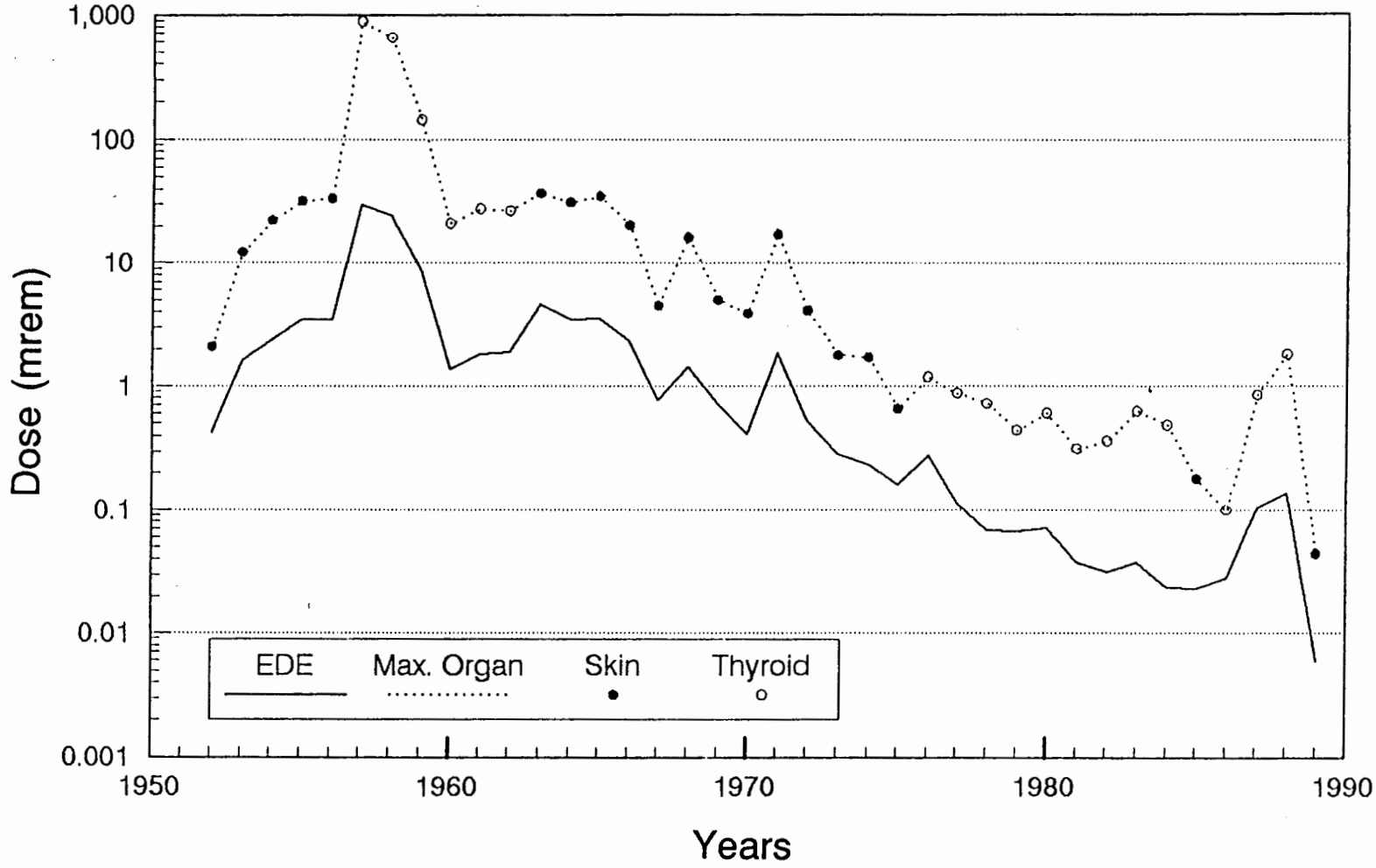


Figure 4–2. Maximally exposed infant operational dose estimates.

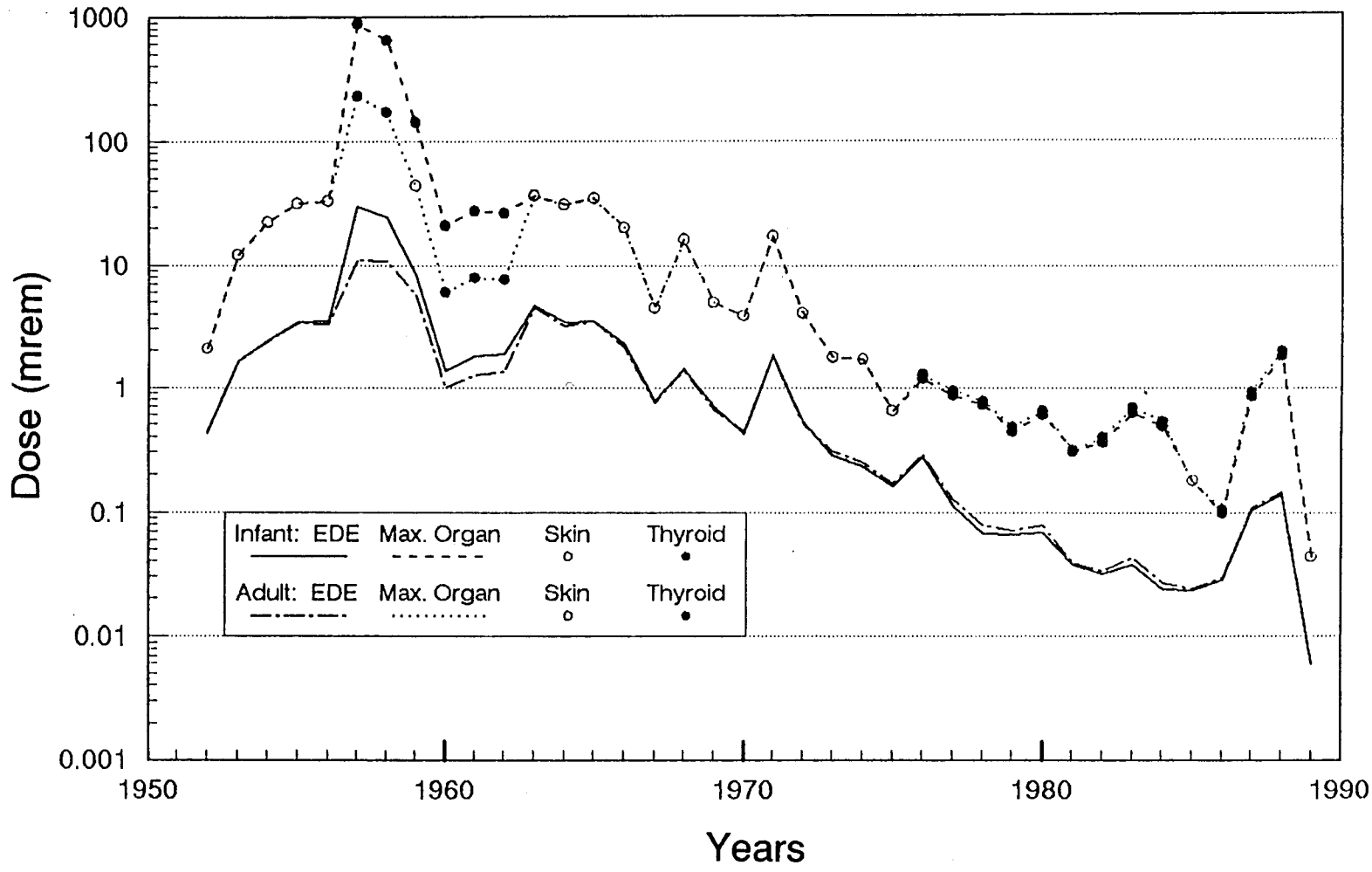


Figure 4–3. Comparison of infant and adult operational dose estimates.

release, the highest EDE comes from zirconium–95, while the highest skin dose is from strontium–90. Also important are cesium–137, cerium–144, yttrium–91, and strontium–89. For Fuel Element Burn Test, B, the highest EDE is due to cesium–137, and the highest skin dose is from strontium–90. Cerium–144 is also an important contributor.

Figure 4–4 shows the total annual doses estimated from episodic and operational releases for each year for the infant EDE. As can be seen, the episodic releases add no more than 1% to the operational releases except in 1955, 1956, 1958, 1959, 1960, and 1961.

The largest EDE from a single release was for IET #3 in 1956. This release accounted for almost 90% of the estimated EDE to the infant during the year. Because meteorological conditions during the IET #3 release were very different from average annual conditions, this release also resulted in a change in the location of maximum exposure during 1956. The operational doses given in Table 4–1 for 1956 are for individuals located in Atomic City. The episodic doses given in Table 4–2 for IET #3 and the combined doses given in Table 4–3 for 1956 are for individuals located at "Building." The procedure for determining the location of largest total dose is given in Section C.3. After IET #3, the three episodic releases that gave the largest offsite dose were the Fuel Element Burn Test, B in 1957, with an infant EDE of 6 mrem (about 13% of the annual total); FPFRT–8 in 1957, with an infant EDE of 17 mrem (about 38% of the annual total); and the SL–1 Accident in 1961, with an infant EDE of 12 mrem (about 87% of the annual total).

Actual radiation doses from the SL–1 Accident are likely to be lower than those estimated in this evaluation. The largest estimated doses from the SL–1 Accident were from iodine–131 through the ingestion pathway. These estimates were made for Atomic City because it was the location having the highest air concentration from SL–1 Accident releases. However, at the time of the accident, the nearest dairy cow was located at Taber, where air concentrations were much lower.

As shown in Table 4–3, the estimated radiation doses were within applicable radiation standards and guidelines in effect at that time for every year of INEL operation. In general, potential offsite doses from INEL activities have been small. The largest radiation doses were calculated for an infant in 1956, when the EDE from operational and episodic releases was estimated to be 61 mrem, and the thyroid dose equivalent was 1350 mrem. Doses from operational and episodic releases during the late 1950s may have been as high as 9% of the applicable whole body standard in effect at that time and 90% of the standard for doses to any single organ of the body.

The trend in recent years has been toward smaller doses as total INEL Site releases have declined. Even with the more restrictive standards in force from 1985 to 1989, offsite radiation doses to a hypothetical individual for those 5 years have been only about 1% of the whole body dose standard and less than 3% of the organ dose standard. Episodic releases have made a substantial contribution to the total potential radiation dose only for a few years during the test programs of the 1950s and early 1960s.

While this study has tended toward conservative dose estimates, there are uncertainties in the estimates of amounts of radionuclides released and transported offsite, particularly for early years of Site operation. Additional uncertainties exist in the evaluation of radiation doses from the four exposure pathways. The uncertainties associated with the highest organ doses (e.g., infant thyroid doses in 1956) may be large enough to preclude the conclusion that no individual exceeded the radiation protection guidelines of 1500 mrem/yr to an organ. Further effort to quantify and evaluate the uncertainties in the calculated doses are beyond the scope of this report. A general discussion of uncertainties is provided in Appendix D.

An environmental surveillance program has been in operation since the early days of INEL

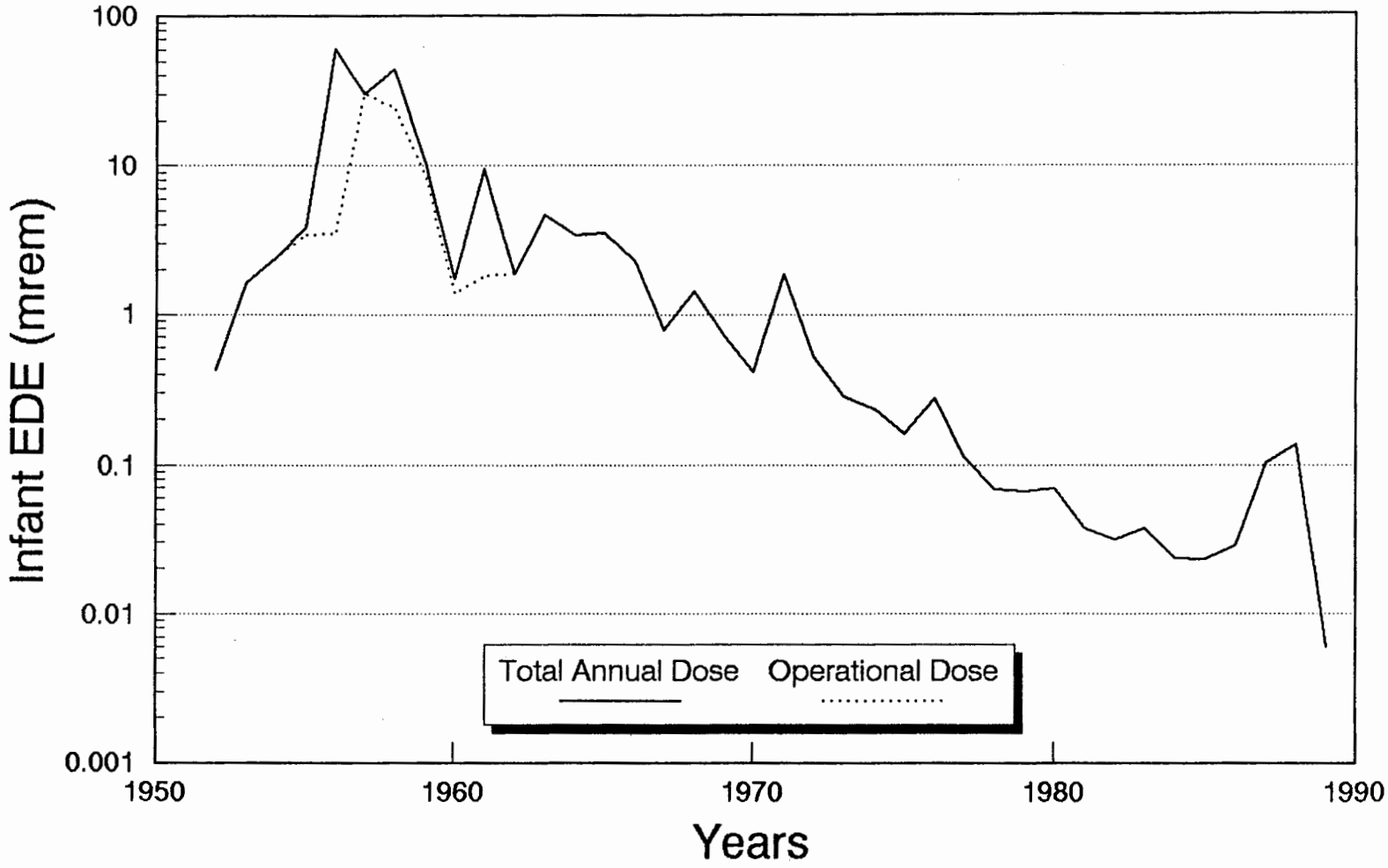


Figure 4–4. Annual operational and episodic dose estimates (infant EDE).

Site operations. Extensive onsite and offsite sampling and measurements have been performed. As the quantities of radioactive materials released from the INEL have declined, the sensitivity of the analyses used in environmental surveillance have improved. The present environmental surveillance program, the history of the program, and a discussion of the results are included in Appendix E.

The estimated radiation doses calculated for this report are not inconsistent with the environmental surveillance results.

4.3 Significance of Radiation Doses

It is important to compare the dose estimates in Tables 4–1, 4–2, and 4–3 to the regulatory standards applicable to each year of INEL Site operation. This comparison shows that estimated radiation doses were within applicable regulations and standards of good practice even as those standards have become more restrictive. A detailed discussion of the evolution of applicable radiation protection standards, dosimetric units, and DOE regulatory guidance is presented in Appendix F.

The radiation protection standards and regulations shown in Table 4–4 are those applicable at the INEL Site boundary for the last year covered by this evaluation. Those standards that apply to all exposure pathways were adopted by DOE in 1985 (Vaughn, 1985; Tseng, 1987). In addition, the EPA issued regulations for airborne radionuclide emissions from DOE facilities that also became effective in 1985 (CFR, 1988). Because of the location and environmental characteristics of the INEL Site, airborne emissions present the dominant pathway for offsite dose. Therefore, the doses calculated in this report from 1985 through the present should be compared against the more restrictive standard of 25 mrem/yr to the whole body and 75 mrem/yr to individual organs. In December 1989, the airborne emission standard for DOE facilities was reduced to 10 mrem/yr and expressed in terms of EDE rather than whole body dose equivalent.

The emission standards apply to routine activities and not to accidents. Protective Action Guides have been established for accidents. The Protective Action Guides are numerically much higher than the emission standards and define the levels where action should be initiated following an accidental release.

A summary of relevant radiation protection standards for the period from 1961 to 1985 is presented in Table 4–5. During this time there

Table 4–4. Radiation protection standards for the protection of the public (1985 to 1989)[a]

All Pathways	Annual Effective Dose Equivalent[b] (mrem)
Occasional annual exposures	500
Prolonged period of exposure[c]	100
Emissions to Air Only[d]	**Annual Dose Equivalent (mrem)**
Whole body	25
Any organ	75

a. Based on dose to individuals at points of maximum probable exposure.

b. The EDE for any member of the public from all routine DOE operations (natural background and medical exposures excluded) shall not exceed these values. Routine operations means normal planned operations and does not include actual or potential accidental or unplanned releases.

c. A prolonged exposure is one that lasts, or is predicted to last, longer than 5 years.

d. Limits of 40 CFR 61, Subpart H, established February 5, 1985, by the EPA. In December 1989, the standard was reduced to 10 mrem/yr EDE, beginning in calendar year 1990.

Table 4–5. Historical radiation protection standards for members of the public (1961 through 1985)[a]

Type of Exposure	Annual Dose Equivalent or Dose Commitment: Based on Dose to Individuals at Points of Maximum Probable Exposure (mrem)	Annual Dose Equivalent or Dose Commitment: Based on Average Dose to a Suitable Sample of the Exposed Population[b] (mrem)
Whole body, gonads, or bone marrow	500	170
Other organs	1500	500

a. Sources for regulations:

1984 – 1981 DOE 5480.1 Chg 2, April 1981 (DOE, 1981)

1980 – 1970 ERDA Manual Chapter 0524, March 1977 (ERDA, 1977b)

1976 – 1963 AEC Manual Chapter 0524, November 1968 (AEC, 1968) and August 1963 (AEC, 1963)

1962 – 1961 Idaho Operations Office, AEC, Announcement No. 163, December 1961 (AEC, 1961)

b. See Paragraph 5.4, FRC Report No. 1 for a discussion of the concept of a suitable sample of the exposed population (FRC, 1960).

was a series of organizational changes within what is now DOE and a continuous development of the science of radiation dosimetry. The basic standards, however, remained essentially unchanged (DOE, 1981; ERDA, 1977b; AEC, 1968; AEC, 1961). The limits applicable to operations at the INEL for maximally exposed members of the public were 500 mrem/yr to the whole body, including blood–forming organs and the gonads, and 1500 mrem/yr to other individual organs.

During the period from 1952 to 1960, radiation protection standards for the general public were under development and for some time consisted of a series of scientific recommendations and standards of good practice rather than clearly defined regulatory limits (FRC, 1960). AEC Manual Chapter 0524, dated February 1, 1958 (AEC, 1958), adopted a preliminary statement by the National Committee on Radiation Protection entitled "Maximum Permissible Radiation Exposures to Man" (NBS, 1957). The preliminary statement was extended and replaced on April 15, 1958, with the publication of an addendum to the National Bureau of Standards Handbook 59 (DOC, 1958). The standards applicable to operations at the INEL for the whole body were 500 mrem/yr from 1957 to 1960 and 1500 mrem/yr from 1952 to 1956. The standards for exposure of the critical organ from 1958 to 1960 were 3000 mrem/yr to the thyroid, skin, and bone and 1500 mrem/yr to other organs. From 1952 to 1957, the standards for exposure to the critical organ can be inferred to be 2500 mrem/yr for the skin and 1500 mrem/yr for other organs. For a more complete discussion, see Appendix F.

Estimated radiation doses were within applicable standards for every year of INEL operation. Table 4–3 shows that offsite doses from operational and episodic releases during the late 1950s may have been as high as 9% of the whole body

dose standard and 90% of the organ dose standard. Even for the more restrictive standards in force from 1985 to 1989, offsite doses to a maximally exposed individual have been only about 1% of the whole body dose standard and less than 3% of the organ dose standard.

In addition to comparing offsite doses to regulatory standards, it is also useful to compare them to the levels of natural background radiation in the vicinity of the INEL. Table 4–6 summarizes the estimated annual dose equivalent from natural sources for an individual living on the Snake River Plain.

Table 4–6. Estimated natural background radiation dose for the Snake River Plain

Source	Annual Effective Dose Equivalent (mrem)
External	
Terrestrial[a]	75
Cosmic[b]	39
Subtotal	114
Internal	
K–40 and others	40
Inhaled nuclides[c]	200
Subtotal	240
Total	354

a. The terrestrial dose for the INEL Site vicinity is based upon soil sampling for natural radionuclides in 1976 (DOE, 1989; ERDA, 1977c).

b. The cosmic dose is derived from the estimate in NCRP Report No. 93 (NCRP, 1987b).

c. The dose from inhaled radionuclides is primarily due to short–lived decay products from radon and varies widely with geographic location. The value shown represents the U. S. population average (NCRP, 1987b).

Radiation from underlying rock is termed the terrestrial contribution to background. This value varies from year to year depending on the amount of shielding provided by snow cover (NCRP, 1987a). In 1988, for example, the light snow cover provided only about a 1% dose reduction, and the terrestrial contribution was estimated to be approximately 75 mrem.

The contribution of extraterrestrial cosmic rays to natural background radiation varies primarily with altitude. The average annual dose equivalent of approximately 26 mrem at sea level doubles with each 2000 m (6560 ft) increase in altitude. The Snake River Plain lies approximately 1500 m (4900 ft) above sea level, resulting in a cosmic dose contribution of about 39 mrem/yr. The sum of the estimated terrestrial and cosmic dose contributions is 114 mrem/yr, which is in close agreement with the average of 107 mrem measured by the Radiological and Environmental Sciences Laboratory for distant and boundary community locations during 1988 (DOE, 1989).

People also receive background radiation from naturally occurring radionuclides deposited within their bodies. One component of this internal dose comes from potassium–40 and other radionuclides incorporated within tissues that contribute about 40 mrem/yr to the total. The most variable contribution is that of inhaled radionuclides, including the short–lived decay products of radon (NCRP, 1987a). The amount of radon in buildings and ground water depends on many factors that may be highly variable, including the radionuclide content of soil and rock in the area. Human exposure to radon also depends on building construction, ventilation, and other factors. Because no specific estimates have been made for Southeastern Idaho, and measurements for homes in this area are few, the U. S. average of 200 mrem/yr appears in Table 4–5 for this component of total background dose. The actual dose for individual residents in the vicinity of the INEL may vary substantially from this value. Thus, the estimated annual EDE from natural

background radiation for a person living in Southeastern Idaho is about 350 mrem.

Doses from airborne releases over the operating history of the INEL have been small compared to doses from sources of natural background radiation, a maximum of 17% of the natural background EDE in 1956. Doses since the early 1970s have been small even when compared to the variability in natural background from year to year and location to location within Southeastern Idaho.

5. QUALITY ASSURANCE

Quality assurance (QA) activities were conducted to ensure that the dose estimates contained in this report were thoroughly documented and defensible (see Appendix G). A QA plan was developed at the beginning of the revision of the draft report. Tasks were identified and individual responsibilities assigned for each of the major steps in the INEL Historical Dose Evaluation. All tasks were reviewed by an individual not responsible for performing those tasks. The quality reviewer examined each task for one or more of the following, as appropriate: (a) adequate documentation, (b) appropriateness of data and assumptions, (c) accurate transcription of data, and (d) verification of calculations.

The Task Group met weekly to discuss problems, report status of individual tasks, and make decisions on important approaches and methods. When the calculations were completed, each member of the Task Group carefully reviewed the draft report. The accuracy of calculations was checked frequently and in depth.

Extensive QA files have been prepared to document the calculations and results appearing in this report. In addition, the INEL Technical Library or the INEL Public Reading Room contain copies of the documents referenced in this report.

6. GLOSSARY

As Low As Reasonably Achievable. An approach to radiation protection to control or manage exposures (both individual and collective to the workforce and general public) as low as social, technical, economic, practical, and public policy considerations permit.

Dose Terms

Absorbed Dose. The energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest in that material. The absorbed dose is expressed in units of rad (or gray) (1 rad = 0.01 gray).

Dose Equivalent. The product of absorbed dose in rads (or gray) in tissue, a quality factor, and other modifying factors. Dose equivalent is expressed in units of rem.

Effective Dose Equivalent. The sum over specified tissues of the products of the dose equivalent in a tissue and the weighting factor for that tissue. The effective dose equivalent is expressed in units of rem. It is the sum of the dose from external sources and the committed effective dose equivalent from internally deposited radionuclides.

Committed Dose Equivalent. The calculated dose equivalent projected to be received by a tissue or organ over a 50–year period after intake of radionuclide into the body. It does not include contributions from external dose. Committed dose equivalent is expressed in units of rem.

Committed Effective Dose Equivalent. The sum of the committed dose equivalents to various tissues in the body, each multiplied by its weighting factor. It does not include contributions from external dose. Committed effective dose equivalent is expressed in units of rem.

Weighting Factor. It is used in the calculation of annual and committed effective dose equivalent to equate the risk arising from the irradiation of tissue to the total risk when the whole body is uniformly irradiated. The weighting factors as defined in ICRP Publication 26 and NCRP Report 91 are shown in the following table:

Organs or Tissues	Weighting Factor
Gonads	0.25
Breasts	0.15
Red Bone Marrow	0.12
Lungs	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder[a]	0.30

a. Remainder means the five other organs or tissue with the highest dose (e.g., liver, kidney, spleen, thymus, adrenal, pancreas, stomach, small intestine, upper large intestine, or lower large intestine). The weighting factor for each remainder organ or tissue is 0.06. The extremities, skin, and lens of the eye are excluded from the remainder organs or tissue for assessment of effective dose equivalent.

Effluent Monitoring is the collection and analysis of samples or measurements of liquid and gaseous effluents for the purpose of characterizing and quantifying contaminants, assessing radiation exposures of members of the public, providing a means to control effluents at or near the point of discharge, and demonstrating compliance with applicable standards and permit requirements.

Environmental monitoring is the collection and analysis of samples or direct measurements of environmental media. Environmental monitoring consists of two major activities: effluent monitoring and environmental surveillance.

Environmental Surveillance is the collection and analysis of samples or direct measurements of air, water, soil, foodstuff, biota, and other media from DOE sites and their environs for the purpose of determining compliance with applicable standards and permit requirements, assessing radiation exposures of members

of the public, and assessing the effects, if any, on the local environment.

Episodic Releases are the tests, experiments, and accidents that took place over a short period of time, typically a few hours.

Operational Releases are continuous, somewhat uniform releases occurring over a year, or over a portion of a year that spans a variety of meteorological conditions.

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Idaho National Engineering Laboratory Historical Dose Evaluation

Volume 2: Appendices

Idaho National Engineering Laboratory

U.S. Department of Energy • Idaho Operations Office

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Vol 2

IDAHO NATIONAL ENGINEERING LABORATORY HISTORICAL DOSE EVALUATION

VOLUME 2: APPENDICES

August 1991

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ABSTRACT

The methodology and results are presented for an evaluation of potential radiation doses to a hypothetical individual who may have resided at an offsite location with the highest concentration of airborne radionuclides near the Idaho National Engineering Laboratory (INEL). Volume 1 contains a summary of methods and results. Volume 2 contains more detailed discussions of methods, data, results, assumptions, and citations of reports and reference material. The years of INEL operations from 1952 to 1989 were evaluated. Radiation doses to an adult, child, and infant were estimated for both operational (annual) and episodic (short-term) airborne releases from INEL facilities. Atmospheric dispersion of operational releases was modeled using annual average meteorological conditions. Dispersion of episodic releases was generally modeled using actual hourly wind speed and direction data at the time of release.

Estimated radiation doses were within the radiation protection standards applicable at that time for every year of INEL operation. In general, potential offsite doses have been small. Doses from operational and episodic releases during the late 1950s may have been as high as 9% of the applicable whole body dose standard in effect at that time and 90% of the standard for doses to individual organs of the body. In addition, the trend in recent years has been toward smaller doses as total INEL Site releases have declined. For the more restrictive standards in force from 1985 to 1989, offsite doses to a hypothetical maximally exposed individual have been about 1% of the whole body standard and less than 3% of the organ dose standard. Episodic releases have made a substantial contribution to the total potential radiation dose only for a few years during test programs of the 1950s and early 1960s. The frequency and size of episodic releases has declined since that time.

FOREWORD

The appendices in this volume provide the details of the calculation of doses performed for the INEL Historical Dose Evaluation. They are intended for a reader with a technical background who desires more information than is given in Volume 1. Included in the appendices are the supporting information for Volume 1, the data, references, assumptions, methods, and results. The complete set of dose calculation results is included on microfiche in a pocket in the back of this volume. Examples are given in the text. The date that each dose calculation was performed is shown at the top of each page. After critical review of the dose calculation results, errors were corrected and the dose calculations for the operational or episodic releases were rerun. Therefore, separate sheets for the same release can show different dates.

Several conventions have been adopted for Volume 2 that are different than Volume 1. In general, three significant figures have been retained in all of the data and calculations until the final doses are listed in Volume 1. This does not indicate that the original data were accurate to three significant figures. If that many figures were present in the original data or generated in the calculational process, they were carried through the calculation to the final step where the results were rounded to one or two significant figures. The accuracies of the source terms, dispersion estimates, and dose calculations suggest that one significant figure is appropriate.

There are numerous direct quotations in the appendices. These direct quotations appear indented and single spaced. At times, the style in the original report was different than today's style. The quotations have been included exactly as they appeared in the original reports, errors included. Where there is an obvious error, it has been marked [sic]. Changes in style are not marked (e.g., in older report, the notation for microcuries was uCi rather than the more correct μCi notation used today).

Scientific notation is used in the appendices (e.g., E-12 means x 10^{-12}, or pico as in picocuries). The prefixes and notation denoting powers of 10 are as follows:

pico	E-12	divide by 1,000,000,000,000
nano	E-9	divide by 1,000,000,000
micro	E-6	divide by 1,000,000
milli	E-3	divide by 1,000
mega	E+6	multiply by 1,000,000

Most of the units for dose used in the report are in millirem (mrem). Some of the dose standards are in Roentgen Equivalent Man (rem). A rem is equal to 1000 mrem.

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ACRONYMS

AEC	U. S. Atomic Energy Commission
ANL-W	Argonne National Laboratory-West
ANP	Aircraft Nuclear Propulsion (Program)
ARA	Auxiliary Reactor Area
ATR	Advanced Test Reactor
BOOT	Burn Out One Tube
BORAX	Boiling Water Reactor Experiment
CEDE	Committed Effective Dose Equivalent
CERT	Controlled Environmental Release Test
CERT	Controlled Environmental Radioiodine Test
CFA	Central Facilities Area
CTF	Core Test Facility
DOE	U. S. Department of Energy
EBR	Experimental Breeder Reactor
EDE	Effective Dose Equivalent
EDF	Experimental Dairy Farm
EFS	Experimental Field Station
EPA	U. S. Environmental Protection Agency
ERDA	Energy Research and Development Administration
ETR	Engineering Test Reactor
EXCES	Experimental Cloud Exposure Study
FECF	Fuel Element Cutting Facility
FEET	Fuel Element Effluent Test
FPFRT	Fission Product Field Release Test
GCRE	Gas-Cooled Reactor Experiment
GE	General Electric

GM	Geiger-Mueller
HSL	Health Services Laboratory
HTRE	Heat Transfer Reactor Experiment
ICRP	International Commission on Radiological Protection
ICPP	Idaho Chemical Processing Plant
IET	Initial Engine Test
INEL	Idaho National Engineering Laboratory
ITS	Idaho Testing Station
IUF	Ingestion Uptake Factor
LDDT	Long Distance Diffusion Test
LIME	Limited Melt Experiment
LOFT	Lost of Fluid Test
LPT	Low Power Test (Facility)
MDC	Minimum Detectable Concentration
MDF	Mesocale Dispersion Factor
MFP	Mixed Fission Product
MTR	Materials Testing Reactor
NIST	National Institute of Standards and Technology
NOAA	National Oceanic and Atmospheric Administration
NRC	U. S. Nuclear Regulatory Commission
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
OMRE	Organic-Moderated Reactor Experiment
PBF	Power Burst Facility
RESL	Radiological and Environmental Sciences Laboratory
RDT	Relative Diffusion Test
RSAC-4	Radiological Safety Analysis Computer Program

RWMIS	Radioactive Waste Management Information System
SL-1	Stationary Low-Power Reactor No. 1
SNAPTRAN	SNAP 10A Transient
SPERT	Special Power Excursion Reactor Test
TAN	Test Area North
TLD	Thermoluminescent Dosimeter
TRA	Test Reactor Area
TSF	Technical Services Facility
USGS	U. S. Geological Survey

APPENDIX A

QUANTITY OF RADIONUCLIDES RELEASED

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A.1 INTRODUCTION

The first step in estimating potential offsite doses from airborne emissions is to determine the source term, that is, the amount of radioactivity released from Idaho National Engineering Laboratory (INEL) facilities and available for transport to the INEL Site boundary. The source term is coupled with an atmospheric dispersion calculation that estimates the ground-level concentrations at selected offsite locations. These air concentrations are then used to calculate doses by various pathways.

The source term must include sufficient information to support the dispersion and dose calculations that depend upon it. Ideally this information includes (a) the amount of each radionuclide released; (b) the locations of facilities from which the radionuclides were released; (c) the time period over which each release occurred; and (d) for some radionuclides, the physical and chemical form of the release.

Dose evaluations are necessarily constrained by available historical data that do not always contain the desired information for an ideal estimate of the source term. There are several reasons for this. For example, stack monitoring data depend on the state of the art in detection instruments used during each year of operation and on the instrumentation installed at each facility. Data quality is also influenced by the recordkeeping, reporting, and quality assurance practices of past years. Both instrumentation and data handling practices have evolved dramatically over the 38 years of INEL operation covered by this analysis. Consequently, careful evaluation, sometimes supported by additional calculations, was required to interpret the available data in a manner that is both technically sound and consistent from year to year. Effluents from previous years cannot be remeasured; therefore, reported emissions are the primary source of information on releases. When additional evaluations were necessary to develop a detailed source term, the assumptions, judgments, calculations, and adjustments are described.

This appendix describes the methods used to interpret the available radionuclide release data for both annual operational and short-term episodic releases and presents the source terms used in subsequent dose calculations.

A.2 METHODS

For the purposes of this analysis, atmospheric releases of radioactivity from the INEL are divided into two categories: (1) annual operational releases, which include continuous, somewhat uniform releases occurring over a year, or a portion of a year, that span a variety of meteorological conditions; and (2) episodic releases, which include all significant releases primarily from short-term operations, tests, and other events that must be treated as distinct and not averaged over the year.

The atmospheric dispersion and dose calculations for each category of release must be handled using slightly different methods. Similarly, the source terms for each category are derived differently. These differences reflect both the sources of available data and the methods of analysis used. Most operational release source terms are obtained from annual facility effluent monitoring data. Plant records were consulted for the years from 1952 to 1963 and checked against effluent monitoring data reports for those years. The Environmental Impact Statement for Waste Management Operations was also used as a reference (ERDA, 1977). Data recorded in the Radioactive Waste Management Information System (RWMIS) were the primary source of information from 1964 to 1989 (Litteer and Reagan, 1990). Most episodic releases are not included in the RWMIS data base and have been reconstructed from operational and engineering information using the inventory and release calculation capabilities of the Radiological Safety Analysis Computer Program (RSAC-4) (Wenzel, 1990). The details of source term data and analyses for each category of releases are described in the following sections.

A.2.1 Decay Chain Equilibria

Many radionuclides are transformed to a stable, nonradioactive element when they decay. However, some radionuclides are transformed into another radionuclide that then decays according to its own unique radiological properties. The quantitative relationship between parent and daughter radionuclides is potentially significant to estimating dose and requires careful evaluation. In those instances in which the half-life of the daughter

radionuclide is less than the half-life of the parent radionuclide, an equilibrium condition is established in which the ratio of the activity of the parent and daughter radionuclides approaches a constant value. Depending on the size of the difference between the parent and daughter half-lives, the relationship is referred to as either transient or secular equilibrium. The decay modes for all radionuclides included in the episodic and operational source terms were, therefore, evaluated to determine those instances in which the equilibrium relationship between parent and daughter radionuclides needed to be considered in calculating external dose to an individual. The equilibrium relationships were assumed to apply at the point of release for operational releases. For episodic releases modeled using the RSAC-4 computer code, the equilibrium relationships were assumed to apply at the INEL Site boundary.

The equilibrium relationship was assumed whenever it would have a significant effect on the external dose received by an individual. The initial presence of the daughter radionuclide was not considered for internal intakes. The decay of the parent radionuclide into the daughter was not considered under the following conditions: (a) the half-life of the daughter is more than 10 times the half-life of the parent, (b) the fraction of parent transformations producing the daughter radionuclide was less than 0.05, (c) the parent and daughter radionuclides were both noble gases with half-lives greater than 5 hours, (d) the parent radionuclide had a half-life of greater than 5 hours and decayed into a noble gas, (e) the half-lives of the parent and daughter were less than 5 minutes, and (f) the half-life of the parent and daughter were greater than 30,000 years. In addition, the consequences of the parent and daughter relationships for Kr-89/Rb-89, Sr-92/Y-92, Ru-105/Rh-105, Sb-127/Te-127, Te-131/I-131, Te-134/I-134, Xe-138/Cs-138, and Ba-142/La-142 were reviewed and determined to have no significant impact on the calculated doses for the reported releases; therefore, they were treated independently.

Table A-1 lists those pairs of nuclides assumed to be in equilibrium. The ratios of the parent to daughter activity at equilibrium are also presented in this table and represent the condition where the rate of production of the daughter radionuclide from decay of the parent equals the

Table A-1. Decay chains included in operational and episodic source terms

Parent Radionuclide	Half-Life	Daughter Radionuclide	Half-Life	Equilibrium Ratio
Kr-88	2.8 h	Rb-88	18 min	1.12
Sr-90	29 yr	Y-90	64 h	1.00
Sr-91	9.5 h	Y-91m	50 min	0.629
Zr-95	64 d	Nb-95	35 d	2.19
Zr-97	17 h	Nb-97m	60 s	0.948
Zr-97	17 h	Nb-97	72 min	1.08
Mo-99	66 h	Tc-99m	6.0 h	0.975
Ru-103	39 d	Rh-103m	56 min	0.998
Ru-105	4.4 h	Rh-105m	45 s	0.246
Ru-106	368 d	Rh-106	30 s	1.00
Sb-125	2.8 yr	Te-125m	58 d	0.245
Sb-129	4.4 h	Te-129	70 min	1.13
Te-131m	30 h	Te-131	25 min	0.225
Te-132	78 h	I-132	2.3 h	1.03
Te-133m	55 min	Te-133	12 min	0.168
Cs-137	30 yr	Ba-137m	2.5 min	0.946
Ba-140	13 d	La-140	40 h	1.15
Ce-144	284 d	Pr-144	17 min	0.986
U-235	7.0E+8 yr	Th-231	26 h	1.00
U-238	4.5E+9 yr	Th-234	24 d	1.00
Np-237	2.14E+6 yr	Pa-233	27 d	1.00

rate of loss as a result of decay of the daughter radionuclide. This equilibrium condition is generally reached after approximately 7 half-lives of the daughter radionuclide. The following equations were used to estimate the activity of the parent and daughter radionuclides, and they formed the basis for estimating the equilibrium ratios shown in Table A-1.

$$A_p(t) = A_p(0)\ e^{-\lambda_p t} \tag{A-1}$$

$$A_d(t) = F_{pd}\ A_p(0) \left[\frac{\lambda_d}{\lambda_d - \lambda_p}\right] (e^{-\lambda_p t} - e^{-\lambda_d t}) \tag{A-2}$$

$$A_g = F_{pd}\, F_{dg}\, A_p(0)\ \lambda_d\, \lambda_g\ (a+b+c) \tag{A-3}$$

$$a = \frac{e^{-\lambda_p t}}{(\lambda_d-\lambda_p)\,(\lambda_g-\lambda_p)} \tag{A-4}$$

$$b = \frac{e^{-\lambda_d t}}{(\lambda_p-\lambda_d)\,(\lambda_g-\lambda_d)} \tag{A-5}$$

$$c = \frac{e^{-\lambda_g t}}{(\lambda_p-\lambda_g)\,(\lambda_d-\lambda_g)} \tag{A-6}$$

where $A_p(t)$ is the activity of the parent, $A_d(t)$ is the activity of the daughter, and $A_g(t)$ is the activity of the granddaughter. At time t, F_{pd} is the fraction of the parent transformations producing the daughter; F_{dg} is the fraction of the daughter transformations producing the granddaughter; and λ_p, λ_d, and λ_g are the decay constants for the parent, daughter, and granddaughter, respectively, and are equal to 0.693 divided by the nuclide half-life. These equations assume that only the parent radionuclide is present at time t = 0 and $A_d(0) = A_g(0) = 0$.

For simplicity of presentation, the daughter radionuclides in equilibrium with the parent radionuclide are not explicitly shown in the source term table presented in Section A.3 or in the microfiche at the back of this volume. However, they are understood to be present with the parent radionuclides and are indicated by the notation +D following the symbol for the parent radionuclide.

A.2.2 Operational Releases

A.2.2.1 RWMIS Data Base

The INEL RWMIS was the principal source of information used to evaluate annual operational releases for this analysis. The RWMIS system was developed in 1971 to provide information on airborne releases and liquid and solid radioactive wastes produced, stored, and disposed of at the INEL (Litteer and Reagan, 1990). Data stored in the RWMIS are provided on a monthly basis by INEL facilities releasing airborne radioactivity. Each year a detailed annual waste report is produced and distributed to all RWMIS users, Federal and state agencies, and to the general public as a report entitled *Radioactive Waste Management Information for 19XX and Record-to-Date*, DOE/ID-10054(XX).

Since its implementation in 1971, there has been an ongoing effort to improve the quality of data in the RWMIS (Litteer and Reagan, 1990). As a result, data on airborne releases since 1971 are generally of good quality and reflect a fairly consistent monitoring technology. When the RWMIS was implemented, a review of the radioactive release records before 1971 was performed. At that time it was determined that the airborne release data for the years 1962 through 1970 were sufficiently complete and of adequate quality to be included as they existed into the RWMIS data base. However, during critical reviews of the facility data by the INEL Historical Dose Evaluation Task Group, significant technical problems were discovered with the RWMIS data from 1962 to 1968. As a result, data on releases from the Idaho Chemical Processing Plant (ICPP) and Test Reactor Area (TRA) were extensively revised (see Sections A.2.2.3 and A.2.2.4). The data for airborne releases during the years 1952 through 1961 were initially judged to be too uncertain and incomplete for inclusion in the RWMIS. Although the total amount of radioactivity released in each of these early years of INEL operation were included in the RWMIS, the amounts of individual radionuclides were not always recorded. Therefore, the data for these years required extensive interpretation and supplementary evaluation to develop the detailed source terms required for this dose evaluation.

Small releases from facilities other than TRA, ICPP, Test Area North (TAN), and Argonne National Laboratory-West (ANL-W) were reported before 1970, but they were combined into an "others" category (Osloond, 1970). This includes releases from GCFR (the Gas-Cooled Reactor Experiment), Naval Reactor Facility (NRF), Organic-Moderated Reactor Experiment (OMRE), Stationary Low-Power Reactor No. 1 (SL-1), and Special Power Excursion Reactor Test (SPERT). At no time were the combined releases more than 0.4% of the INEL total releases. The releases from TAN "during 1957 through 1961 are attributed to ANP operations" (Osloond, 1970) and are included the in episodic releases.

A.2.2.2 Adjustments to the RWMIS

A series of adjustments to the RWMIS data base were required to compensate for changes in monitoring technology and reporting practices over the years. In general, there were few adjustments required for recent years and more required for earlier years. The aim of these adjustments was to determine the mixture of radionuclides as accurately as possible and to include all of the radioactivity that could contribute significantly to offsite doses in the source term.

The criteria for including radionuclides in the annual operational source term for each year were as follows:

- All beta-ray and gamma-ray emitting radionuclides, including "unidentified beta activity" released in excess of 1 mCi
- All isotopes of iodine released in excess of 0.1 mCi
- All alpha emitting radionuclides, including "unidentified alpha activity" released in excess of 0.01 mCi
- Any reported releases of Sr-90, Cs-137, or isotopes of plutonium.

Radionuclides excluded from the operational source term using these criteria would add no more than 0.0001 mrem to the effective dose equivalent (EDE). In addition to these screening criteria, some adjustments to the RWMIS

data were required to provide an accurate list of radionuclides sufficient to support dose calculations. There were three principal types of adjustments. The first, and most common adjustment, accounted for the production of radionuclides (daughter nuclides) that are not necessarily included in the monitoring data but result from the decay of other radionuclides (parent nuclides) present in the release. The second type of adjustment estimated the specific mix of radionuclides included in reported releases of gross amounts of radioactivity such as "unidentified beta plus gamma" (UN-ID-B+G) and "unidentified alpha" (UN-ID-ALPHA) activity. The third type of adjustment was used to estimate the amount of each radionuclide present in those cases where the total activities of two nuclides were reported together because the analytical instrumentation of the day could not resolve them. Sections A.2.2.2.1 through A.2.2.2.3 elaborate on these three adjustment types.

A.2.2.2.1 Decay Chain Equilibria. The decay chain equilibria adjustments were made to account for the production of radionuclides that are not necessarily included in the monitoring data but result from the decay of other radionuclides present in the release. Many radionuclides are transformed to a stable, nonradioactive element when they decay. However, some radionuclides are transformed into another radionuclide, which then decays according to its own radiological properties. The quantitative relationship between parent and daughter radionuclides is addressed in Section A.2.1.

When only the parent radionuclide appeared in the RWMIS data base, the daughter was also assumed to have been produced and to be present in the equilibrium ratio. For some pairs of radionuclides both the parent and daughter appear in the RWMIS data base. In such cases, the amount of the daughter expected from assuming equilibrium is included with the parent nuclide and indicated by the notation +D in Section A.3. If the reported activity of the daughter nuclide is less than that expected from equilibrium, the assumption that equilibrium exists results in a conservative overestimate of the release. If the reported activity of the daughter exceeds that expected from equilibrium conditions, then the excess is listed explicitly in the table and included separately in subsequent dose calculations.

The following detailed discussions contain additional information on RWMIS entries that required decay chain equilibria adjustments. The adjustment discussions are ordered from the most to the least common. They apply primarily to the years from 1964 through 1989. Discussions of the assumptions made for 1952 to 1968 for TRA and 1953 to 1963 for ICPP source terms are contained in Sections A.2.2.3 and A.2.2.4.

Kr-88 and Rb-88: The most common adjustment to the RWMIS data was to include the contribution from decay of Kr-88 to Rb-88. This adjustment was required for all years between 1969 and 1989. While Kr-88 releases were reported for all those years, small quantities of Rb-88, much less than expected from assuming equilibrium, were reported for most of these years. Because of the short half-life of Rb-88 (18 minutes), decay equilibrium will generally be established by the time emitted radionuclides reach the INEL Site boundary.

Zr-95 and Nb-95: For the years 1977, 1976, 1974, 1973, 1972, and 1971 these two radionuclides were reported separately in the RWMIS. For most of these years the assumption of equilibrium appears to be accurate or slightly conservative (see Table A-2). However, in 1973 the amount of Nb-95 reported in the RWMIS report significantly exceeded the amount expected from equilibrium, demonstrating that equilibrium conditions did not exist. For this year, the Nb-95 in excess of expected equilibrium activity is listed separately in the source term.

The Zr-95 used in the source term for 1971 includes that activity listed in RWMIS as Zr-95 and a small additional amount equal to the portion of the unresolved Zr-Nb-95, which was assumed to be Zr-95. For a more complete discussion of activity reported as Zr-Nb-95 see Section A.2.2.2.3.

Te-132 and I-132: Releases of Te-132 and I-132 appear in the RWMIS only for 1967. Because of differences in the chemical properties of these two elements, it is possible for a larger fraction of the iodine inventory in a facility to be released than the tellurium inventory. For 1967, the release of Te-132 was 0.39 Ci, and the release of I-132 was 0.46 Ci. The 0.39 Ci of I-132 expected from decay equilibrium was included with the parent in the

Table A-2. Evaluation of reported Zr-95 and Nb-95 releases

	Amount of Radioactivity (Ci)		
Year	Zr-95 Reported in the RWMIS	Nb-95 Expected from Equilibrium	Nb-95 Reported in the RWMIS
1977	0.0017	0.0037	0.0030
1976	0.0022	0.0047	0.0048
1974	0.0042	0.0092	0.0080
1973	0.023	0.051	0.064
1972	0.33	0.70	0.71
1971	1.7	3.6	3.4

source term and is indicated with a +D. The excess 0.055 Ci of I-132 is listed separately.

Ba-140 and La-140, Ru-106 and Rh-106: Two pairs of parent/daughter radionuclides were not reported separately in the RWMIS but were reported together, presumably in equilibrium. They were originally reported as Ba-La-140 and Ru-Rh-106 in the 1979 and 1965 RWMIS reports, respectively. In these cases adjustments were made so that the source term contains both parent and daughter in equilibrium amounts.

Sr-91 and Np-237: Strontium-91 and Np-237 parent radionuclides were only listed in the RWMIS for 1979 and 1975, respectively. In both cases, equilibrium was conservatively assumed to exist between the parent and daughter radionuclides.

A.2.2.2.2 Unidentified Radioactivity. In some cases monitoring instruments could not identify the specific radionuclides released; instead, they detected gross measures of radioactivity that were reported as UN-ID-B+G or UN-ID-ALPHA. Releases were more commonly reported in these gross terms in earlier years of INEL operation before the development of more sophisticated monitoring technologies. These entries were evaluated by using engineering and operational information specific to each INEL facility and each year of INEL Site operation to estimate the amount of specific radionuclides that would have been present.

The following detailed discussions contain information on the unidentified radioactivity adjustments.

UN-ID-B+G: UN-ID-B+G activity above the screening level was reported for 1971.

The UN-ID-B+G activity was assumed to be a mixture of particulate activity in the same proportions as the long-lived particulate radionuclides identified in the rest of the RWMIS source term for this year. Particulate radionuclides with half-lives less than 1 day were not considered. The total activity of parent and daughter radionuclides was considered when determining the distribution of the UN-ID-B+G. For 1971, the UN-ID-B+G was 0.08% of the total particulate activity released.

UN-ID-ALPHA: UN-ID-ALPHA activity above the screening level was reported for 1969 through 1973 from the ICPP and, in smaller amounts, for 1971 through 1976 from other facilities [Auxiliary Reactor Area (ARA), Power Burst Facility (PBF), TRA, and TAN]. The UN-ID-ALPHA was conservatively assumed to be composed of Pu-238, Pu-239, and Pu-240 in the same mixture as released from the ICPP. Data for estimating the mixture of alpha emitting nuclides were obtained from monitoring data for the 13-year period from 1974 to 1986, when the effluent samples at the ICPP were analyzed specifically for plutonium isotopes. The ratio of Pu-238 to Pu-239/240 ranged from 2.2 to 12 and averaged 6.8. (Pu-239/240 represents the sum of the Pu-239 and Pu-240 activity because these nuclides are difficult to identify separately.) The average ratio of 6.8 was used to approximate the mixture of the UN-ID-ALPHA for 1969 through 1976.

For 1964 through 1968, no alpha emissions were recorded in the RWMIS data base for the ICPP. Although not recorded, alpha emissions (assumed to be entirely plutonium) were likely for these years. To provide an estimate of the plutonium emissions, the ratio of Sr-90 emissions to total plutonium emissions was calculated for 1969 through 1974. The average ratio during this time period was approximately 400. To estimate the total plutonium activity released in 1967 and 1968, the Sr-90 emissions were divided by the ratio. The plutonium activity was assumed to be Pu-238 and Pu-239/240, and the ratio of Pu-238 to Pu-239/240 was assumed to be 6.8.

Pu-TOTAL: The sum of the activity of Pu-238, Pu-239, and Pu-240 released to the air was reported as Pu-TOTAL from 1987 through 1989. The isotopic make up of the Pu-TOTAL was assumed to be similar to the average ratio of Pu-238 to Pu-239/240 during the 13-year period from 1974 to 1986 for the ICPP. The ratio of Pu-238 to Pu-239/240 ranged from 2.2 to 12 and averaged 6.8 during this time period.

I-129: I-129 was not reported in the RWMIS reports until 1979. Amounts of I-129 released in each year before 1979 were estimated based on the amount of I-129 in the fuel that was processed and on studies of when I-129 was released in the processing. Approximately 80% of the I-129 is released as an airborne effluent and 20% is in the liquid effluent. Most of the airborne I-129 was released in the calcination process. The Waste Calcining Facility started operating in 1963. Plant records were consulted to determine the quantities of high-level waste being calcined in each campaign. Before 1963, small quantities of I-129 were released as a result of fuel dissolution. These quantities were estimated using RSAC-4. In addition, investigations at the ICPP have shown that about one-third of the I-129 released is elemental iodine (I_2) or hypoiodous acid (HOI) and two-thirds is organic iodine. The adjustment into elemental and organic forms was made for all years of operation. The distinction is important for the dose calculation because elemental iodine is more readily deposited on pasture grass and food crops than organic iodine.

Mixed Fission Products: Mixed fission products (MFPs) were reported for the ANL facilities during 1964. These MFPs were noble gases released from the Boiling Water Reactor No. 5 (BORAX-V) reactor and were primarily Kr-85 and Xe-135. Because the air immersion dose conversion factors for Xe-135 are greater than those for Kr-85, the release was assumed to be entirely Xe-135.

A.2.2.2.3 Unresolved Radionuclides. Adjustments were made to estimate the mix of radionuclides in those cases where the total activity of the pairs of nuclides had been reported together because the instrumentation of the day could not resolve them. The gamma-ray spectrometry instrumentation available before 1970 was sometimes not able to separate gamma-rays that were very close in energy. Therefore, the results were often reported as the sum of the

activity measured for the energy area of interest. This resulted in the total activity of some pairs of nuclides being reported in the RWMIS. These pairs of radionuclides are discussed below.

Zr-Nb-95: During the period from 1964 through 1971, the activities of Zr-95 and Nb-95 were reported as a sum of the two nuclides. Because Zr-95 is the parent nuclide of Nb-95, equilibrium was assumed to exist between the two. In the source term used for the dose evaluation, the Nb-95 daughter is not explicitly listed but is understood to be present in the Zr-95+D term; the total activity of the parent and daughter is equal to the value reported in RWMIS for Zr-Nb-95.

Ce-141-144: The combined activities of Ce-141 and Ce-144 were reported in the 1964 and 1966 RWMIS. In evaluating source terms for these years, all the activity was assumed to be Ce-144 rather than Ce-141. This assumption is conservative because the dose conversion factors and the half-life for Ce-144 are larger than those for Ce-141.

Ru-103-106: The combined activities of Ru-103 and Ru-106 were reported in the 1964 and 1965 RWMIS. In evaluating source terms for these years, all the activity was assumed to be Ru-106 because the dose conversion factors and half-life for Ru-106 are larger than those for Ru-103.

Sr-89-90: The combined activities of Sr-90 and Sr-89 were reported in the 1965 RWMIS because some radiochemical analyses are unable to distinguish between Sr-89 and Sr-90. In evaluating source terms for this year, the release was conservatively assumed to be entirely Sr-90.

A.2.2.3 1952 Through 1968 TRA Releases

A critical review of the radionuclide airborne effluent data in the RWMIS for the years before 1969 indicated that errors were apparently made. Some of the information in the RWMIS was based on calculations made in the early 1970s. For 1963 through 1966, 1.4 to 1.9% of the yearly release was attributed to an "unidentified beta-gamma" component. For 1962 the effluent data did not contain this component. For the period 1962 through 1968, the

RWMIS reports ascribed from 6 to 41% of the annual release to Xe-137 (AEC, no date), a radionuclide with a short half-life of 3.84 minutes; however, the annual releases are reported as 0 in 1969 and 1970. AEC (1961) shows that Xe-137 amounted to 30% of the 1960 annual release from TRA. For the years where Xe-137 was reported, the quantity of Xe-138, which has a radiological half-life of 14.1 minutes, was always less than the value for Xe-137 with a shorter half-life.

The puzzling aspects of these reports, based on what is currently measured with a state-of-the-art lithium-drifted germanium (GeLi) gamma-ray spectrometer in the Advanced Test Reactor (ATR), are twofold: (1) that Xe-137 was reported at all and (2) that the corresponding quantity of Xe-138 for each of the years was always less than the quantity of Xe-137. The same type of anomaly exists for the reported quantities of Kr-88 and Kr-89. Also, before 1962, the entire amount of airborne effluent was labeled as "unidentified beta-gamma," which is insufficient for dose calculation purposes. The following discussion provides the reasons why the authors believe Xe-137 was erroneously reported, what was done to determine the constituents of "unidentified beta-gamma," and how the anomaly was corrected in the xenon and krypton data.

The noble fission gases, krypton and xenon, are produced at the INEL only in the core region of an operating nuclear reactor. Calculating the equilibrium amounts of the fission gases with the RSAC-4 computer code for the case of a single fuel assembly, operated with parameters applicable to the Materials Testing Reactor (MTR), produces the noble gas inventory given in Table A-3. As shown in this table, the amounts of Kr-88 and Kr-89 and Xe-137 and Xe-138 in the core (at 0.0 decay) are of roughly equal quantities. As the gases leave the region of the core and are carried by the primary coolant to the degassification unit, they decay according to their respective radiological half-lives and according to the time that it takes to travel from the core to the degassifier. This transit time is 5.60 minutes for the MTR (Heath and Passell, 1955). Column 4 of Table A-3 shows the predicted amounts of krypton and xenon isotopes that would be present in the degassifier. It takes another 60 or 70 seconds for the gases to travel to the stack where they would be sampled or measured by the stack gas monitor. As shown in Table A-3,

Table A-3. Fission gas inventory at equilibrium conditions for MTR fuel assembly (calculated by RSAC-4)

Radionuclide	Radiological Half-life	0.0 Decay (Ci)	5.6 Min Decay (Ci)
Kr-85m	4.48 h	1.13E+4	1.13E+4
Kr-85	10.7 yr	1.28E+1	1.28E+1
Kr-87	76.3 min	3.38E+4	3.24E+4
Kr-88	2.84 h	4.79E+4	4.69E+4
Kr-89	3.16 min	6.18E+4	1.83E+4
Kr-90	32.3 s	6.30E+4	4.67E+1
Kr-91	8.57 s	4.55E+4	1.08E-7
Kr-92	1.84 s	1.99E+4	0.0
Kr-93	1.27 s	6.74E+3	0.0
Xe-129m	8.89 d	7.02E-5	7.01E-5
Xe-131m	11.8 d	2.58E+2	2.58E+2
Xe-133m	2.19 d	2.58E+3	2.58E+3
Xe-133	5.24 d	8.62E+4	8.62E+4
Xe-135m	15.4 min	1.61E+4	1.56E+4
Xe-135	9.11 h	8.96E+4	8.96E+4
Xe-137	3.83 min	8.18E+4	3.16E+4
Xe-138	14.1 min	8.30E+4	6.32E+4
Xe-139	39.5 s	6.78E+4	1.88E+2
Xe-140	13.6 s	4.83E+4	1.76E-3
Xe-141	1.72 s	1.57E+4	0.0
Xe-142	1.22 s	5.10E+3	0.0

the quantities of Kr-88 and Xe-138 are respectively greater than the quantities for Kr-89 or Xe-137.

The degassification system for the ATR is similar to that for the MTR. However, the ATR transit time for primary coolant from the core to the degassifier unit is only 45 seconds. Also, there is a real-time GeLi gamma-ray spectrometer located on the gas plenum, immediately downstream of the degassification unit. Therefore, the possibility of detecting Xe-137 in the off-gas of the ATR would be greater than for the MTR. Table A-4 shows the reported airborne releases of radioactive noble gases for the ATR when the reactor was operating at 155 MW. The lower limit of detection for each of the gases that were reported is also presented in the table.

Table A-4. Real-time stack gas monitor data for ATR[a]

Radionuclide	Release Data (Ci/d)	Lower Limit of Detection (Ci/d)
Ar-41	8.14	0.012
Kr-85m	0.03	0.016
Kr-87	0.26	0.04
Kr-88	0.22	0.019
Xe-133 (seldom seen)		0.09
Xe-135	0.19	0.02
Xe-135m	0.14	0.02
Xe-138	0.17	0.10

a. Data for 3/25/89 with reactor power at 155 MW.

Only under very unusual circumstances, when gaseous activity is on the order of 1000 times the ordinary effluent amount, would Xe-137 be identified. For example, the analysis appearing in Heath and Passell (1955) describes the isotopic composition of an MTR gas sample taken at the degassifier for the purpose of characterizing the process water off-gas stream. The report states, "The following fission product gaseous activities were observed either by direct observation or through identification of daughter activities: Kr-85, Kr-87, Kr-88, Xe-133, Xe-135, Xe-138, and Xe-139" (Heath and Passell, 1955, p. 3). There was no mention of Kr-89 or Xe-137 in this report. C. L. Rowsell, who has been involved with identification of airborne effluent components from 1961 to the present, stated that he knows of no instance where Xe-137 has been identified as even a minor component of TRA airborne effluent.[a] In discussion with other personnel involved in the preparation of TRA airborne effluent reports during the 1960s, they indicated that these values were calculated values. However, the authors of this report were unable to discover any documentation relating to the procedures used for these calculations. Production of Xe-137 by the n-gamma reaction with Xe-136 was also investigated and rejected as a possible source.

a. Personal communication between C. L. Rowsell and H. K. Peterson of EG&G Idaho, Inc., April 28, 1989.

Because of these apparent anomalies in the RWMIS airborne effluent report, an extensive review of the MTR and Engineering Test Reactor (ETR) literature was performed to determine if there was a more reliable way of estimating noble gas and "unidentified beta-gamma" releases. It was decided that the individual cycle reports for each facility should be used. These reports record the airborne effluent as measured on a daily basis by continuous stack monitors. The stack monitors were calibrated against weekly or semiweekly gas and particulate samples analyzed by NaI gamma spectrometry.

However, cycle reports showed the total gaseous activity and the total particulate activity without differentiating among the various radionuclides; therefore, additional calculations were required to determine the mixture released. The ETR reported the gaseous activity as "Ar-41" and "other gases" (i.e., Kr-88, Kr-89, and Xe-138). For this facility, the particulate activity was reported as the total of Rb-88, Rb-89, and Cs-138, all relatively short-lived particulate species of radionuclides. In a study performed in 1966 to more precisely define the quantities of airborne effluent components for a companion facility, the MTR, it was determined that Ar-41 was released from the MTR at the rate of 367 Ci per 40 MWd (Peterson, 1966). Using this factor, the MTR gaseous activity was broken down to terms that were consistent with those for the ETR. A detailed breakdown for the two reactors for the years 1960 through 1963 is provided in Table A-5. Also presented for comparison in this table is the RWMIS data for the same years.

The total of all TRA airborne effluent activity for the years 1960 through 1963, based on facility cycle reports, was 1.46E+6 Ci. Of this total, 44.85% was Ar-41; 53.6% was other gaseous activity (e.g., Kr-85 and Xe-133); and 1.55% was particulate activity. To determine the proper fraction of Ar-41, other gases, and particulates for TRA airborne effluents for 1952 through 1968, these percentages were applied to the RWMIS TRA yearly totals. To further subdivide the "gas" and "particulate" into individual radionuclides, the proportions applicable to the ATR (presented in Table A-6) were used. To ensure that all of the activity released is ascribed to a radionuclide for dose calculation purposes, "gross beta" is conservatively assumed to be Sr-90, and "gross alpha" is conservatively assumed to be Pu-239. Based on the review of the facility documentation, the values for the

Table A-5. ETR and MTR radioactive airborne effluent as extracted from the facility cycle reports

1960	Ar-41 (Ci)	Other Gas (Ci)	Particulate (Ci)
ETR	(62,900)[a]	52,600	19,800
MTR	99,700[b]	50,900	11.8
Total 1960 TRA Airborne Effluent = 285,900 Ci; (RWMIS = 232,500 Ci)			
1961			
ETR	113,800	9,900	480
MTR	103,900[b]	104,600	13.8
Total 1961 TRA Airborne Effluent = 332,700 Ci; (RWMIS = 335,400 Ci)			
1962			
ETR	47,900	53,600	1,920
MTR	103,000[b]	111,400	14.4
Total 1962 TRA Airborne Effluent = 317,800 Ci; (RWMIS = 349,100 Ci)			
1963			
ETR	27,200	11,900	364
MTR	97,500[b]	388,800	23.1
Total 1963 TRA Airborne Effluent = 525,800 Ci; (RWMIS = 598,700 Ci)			

a. Not reported in cycle reports; value within parentheses is average for ETR Ar-41 for 1961 through 1963.

b. Argon-41 quantity for MTR is calculated by multiplying the megawatt-days that MTR generated by 9.175.

Table A-6. Isotopic composition of 1987 ATR airborne effluent used to reconstruct TRA releases

Radionuclide	Ci	Component (%)
Ar-41	2.40E+3	---
Other Gaseous Components		
Kr-85	2.99E+0	4.00E-1
Kr-85m	2.79E+1	3.78E+0
Kr-87	1.01E+2	1.37E+1
Kr-88	9.89E+1	1.34E+1
Xe-133	1.08E+1	1.46E+0
Xe-135	1.07E+2	1.45E+1
Xe-135m	5.90E+1	7.99E+0
Xe-138	3.31E+2	4.48E+1
Particulate and Iodine Components		
Rb-88	1.30E-1	2.40E+1
Rb-89	2.05E-1	3.79E+1
Cs-138	1.85E-1	3.42E+1
Ba-139	1.75E-2	3.23E+0
Na-24	1.67E-3	3.09E-1
Hg-203	6.53E-4	1.21E-1
Tc-99m	5.10E-4	9.42E-2
Y-91m	2.29E-4	4.23E-2
Co-60	4.40E-5	8.13E-3
Cr-51	5.77E-5	1.07E-3
Cs-134	3.14E-5	5.80E-3
Cs-137	7.80E-5	1.44E-2
I-131	8.49E-5	1.57E-2
I-133	6.74E-5	1.25E-2
Sr-90	6.45E-5	1.19E-2
Ba-140	6.01E-6	1.11E-3
La-140	9.86E-6	1.82E-3
Ce-141	3.82E-7	7.06E-5
Gross beta	2.62E-4	4.84E-2
Gross alpha	4.28E-8	7.91E-6

particulate components assigned in this manner will conservatively bound the actual release quantities. The resulting detailed reconstruction of radionuclide emissions from the TRA from 1968 to 1952 is shown in Tables A-7 through A-11. Adjustments for decay chain equilibria similar to those described in Section A.2.2.2.1 were made for Ba-140 and La-140 and for Kr-88 and Rb-88 before addition into the source terms in Section A.3.

A.2.2.4 1953 Through 1963 ICPP Releases

Airborne releases from the ICPP for 1953 through 1963 were reconstructed to calculate all of the radionuclides that would have been discharged to the environment. Because of the lack of information available for the years before 1957 on the operation of the ICPP, data and information developed in later years were used to reconstruct releases for the early years of plant operation. The reconstruction began with the Radioactive Lanthanum (RaLa) operations as described below.

A.2.2.4.1 RaLa Process Operations at the ICPP (February 1957 to April 1963). From February 1957 to April 1963, spent fuel elements were processed at the ICPP to recover the fission product Ba-140 and the residual U-235. When Ba-140 undergoes radioactive decay, radioactive lanthanum-140 results. When the La-140 radionuclide decays, it emits a high-energy gamma ray that was desired for a special need. Although the ICPP RaLa process resulted in recovering kilocurie quantities of Ba-140, it took its name from the desired end product.

Usually the fuel processing at the ICPP is performed only after the fuel has been out of the reactor for at least 120 days, allowing for significant decay of the short-lived radioactive iodine fission products (ERDA, 1977). Because Ba-140 has a radioactive half-life of only 12.8 days, the fuel was processed as soon as possible after being removed from the reactor (normally about 2 to 7 days) (Legler et al., 1957). This resulted in relatively large airborne releases of short-lived radioactivity, particularly noble gases and iodines. Releases of I-131 were of special concern and higher than expected from the design studies. During the early RaLa campaigns, improvements were

Table A-7. Reconstructed TRA airborne effluent for the years 1968, 1967, 1966, 1965, and 1964 (in Ci)

Radionuclide	1968	1967	1966	1965	1964
Gaseous Component					
Ar-41	5.66E+4	8.89E+4	1.50E+5	2.34E+5	2.55E+5
Kr-85	2.70E+2	4.26E+2	7.16E+2	1.12E+3	1.22E+3
Kr-85m	2.56E+3	3.99E+3	6.77E+3	1.05E+4	1.15E+4
Kr-87	9.25E+3	1.45E+4	2.45E+4	3.81E+4	4.16E+4
Kr-88	9.06E+3	1.42E+4	2.40E+4	3.74E+4	4.08E+4
Xe-133	9.89E+2	1.55E+3	2.62E+3	4.07E+3	4.44E+3
Xe-135	9.82E+3	1.53E+4	2.60E+4	4.04E+4	4.41E+4
Xe-135m	5.40E+3	8.47E+3	1.43E+4	2.23E+4	2.43E+4
Xe-138	3.03E+4	4.75E+4	8.03E+4	1.25E+5	1.36E+5
Particulate and Iodine Component					
Rb-88	4.68E+2	7.37E+2	1.24E+3	1.94E+3	2.12E+3
Rb-89	7.40E+2	1.16E+3	1.96E+3	3.06E+3	3.34E+3
Cs-138	6.68E+2	1.04E+3	1.77E+3	2.76E+3	3.01E+3
Ba-139	6.31E+1	9.92E+1	1.67E+2	2.61E+2	2.85E+2
Na-24	6.04E+0	9.46E+0	1.60E+1	2.49E+1	2.72E+1
Hg-203	2.37E+0	3.71E+0	6.27E+0	9.76E+0	1.06E+1
Tc-99m	1.84E+0	2.89E+0	4.88E+0	7.60E+0	8.31E+0
Y-91m	8.27E-1	1.30E+0	2.19E+0	3.41E+0	3.73E+0
Co-60	1.59E-1	2.49E-1	4.21E-1	6.56E-1	7.17E-1
Cr-51	2.09E-2	3.28E-2	5.54E-2	8.63E-2	9.44E-2
Cs-134	1.13E-1	1.78E-1	3.00E-1	4.68E-1	5.12E-1
Cs-137	2.82E-1	4.41E-1	7.46E-1	1.16E+0	1.27E+0
I-131	3.07E-1	4.83E-1	8.13E-1	1.27E+0	1.39E+0
I-133	2.44E-1	3.84E-1	6.47E-1	1.01E+0	1.10E+0
Sr-90	1.18E+0	1.85E+0	3.12E+0	4.87E+0	5.32E+0
Ba-140	2.17E-2	3.40E-2	5.75E-2	8.96E-2	9.79E-2
La-140	3.56E-2	5.59E-2	9.43E-2	1.47E-1	1.61E-1
Ce-141	1.38E-3	2.17E-3	3.66E-3	5.70E-3	6.23E-3
Pu-239	1.55E-4	2.42E-4	4.10E-4	6.38E-4	6.98E-4

Table A-8. Reconstructed TRA airborne effluent for the years 1963, 1962, and 1961 (in Ci)

Radionuclide	1963	1962	1961
Gaseous Component			
Ar-41	2.69E+5	1.57E+5	1.50E+5
Kr-85	1.28E+3	7.49E+2	7.19E+2
Kr-85m	1.21E+4	7.07E+3	6.80E+3
Kr-87	4.39E+4	2.56E+4	2.46E+4
Kr-88	4.30E+4	2.51E+4	2.41E+4
Xe-133	4.69E+3	2.73E+3	2.62E+3
Xe-135	4.65E+4	2.71E+4	2.60E+4
Xe-135m	2.56E+4	1.50E+4	1.44E+4
Xe-138	1.44E+5	8.39E+4	8.06E+4
Particulate and Iodine Component			
Rb-88	2.23E+3	1.30E+3	1.25E+3
Rb-89	3.51E+3	2.05E+3	1.97E+3
Cs-138	3.17E+3	1.85E+3	1.78E+3
Ba-139	3.00E+2	1.75E+2	1.68E+2
Na-24	2.87E+1	1.67E+1	1.61E+1
Hg-203	1.12E+1	6.55E+0	6.29E+0
Tc-99m	8.74E+0	5.10E+0	4.90E+0
Y-91m	3.93E+0	2.29E+0	2.20E+0
Co-60	7.54E-1	4.40E-1	4.23E-1
Cr-51	9.93E-2	5.79E-2	5.56E-2
Cs-134	5.38E-1	3.14E-1	3.02E-1
Cs-137	1.34E+0	7.79E-1	7.49E-1
I-131	1.46E+0	8.50E-1	8.16E-1
I-133	1.16E+0	6.76E-1	6.50E-1
Sr-90	5.60E+0	3.26E+0	3.13E+0
Ba-140	1.03E-1	6.01E-2	5.77E-2
La-140	1.69E-1	9.85E-2	9.46E-2
Ce-141	6.55E-3	3.82E-3	3.67E-3
Pu-239	7.34E-4	4.28E-4	4.11E-4

Table A-9. Reconstructed TRA airborne effluent for the years 1960, 1959, and 1958 (in Ci)

Radionuclide	1960	1959	1958
Gaseous Component			
Ar-41	1.04E+5	7.58E+4	4.65E+4
Kr-85	4.98E+2	3.62E+2	2.22E+2
Kr-85m	4.71E+3	3.42E+3	2.10E+3
Kr-87	1.70E+4	1.24E+4	7.59E+3
Kr-88	1.67E+4	1.21E+4	7.44E+3
Xe-133	1.82E+3	1.32E+3	8.11E+2
Xe-135	1.81E+4	1.31E+4	8.05E+3
Xe-135m	9.96E+3	7.23E+3	4.44E+3
Xe-138	5.59E+4	4.06E+4	2.49E+4
Particulate and Iodine Component			
Rb-88	8.66E+2	6.29E+2	3.86E+2
Rb-89	1.36E+3	9.91E+2	6.08E+2
Cs-138	1.23E+3	8.95E+2	5.49E+2
Ba-139	1.16E+2	8.46E+1	5.19E+1
Na-24	1.11E+1	8.09E+0	4.96E+0
Hg-203	4.36E+0	3.17E+0	1.94E+0
Tc-99m	3.39E+0	2.47E+0	1.51E+0
Y-91m	1.52E+0	1.11E+0	6.79E-1
Co-60	2.93E-1	2.13E-1	1.31E-1
Cr-51	3.86E-2	2.80E-2	1.72E-2
Cs-134	2.09E-1	1.52E-1	9.31E-2
Cs-137	5.19E-1	3.77E-1	2.31E-1
I-131	5.66E-1	4.11E-1	2.52E-1
I-133	4.50E-1	3.27E-1	2.01E-1
Sr-90	2.17E+0	1.58E+0	9.68E-1
Ba-140	4.00E-2	2.91E-2	1.78E-2
La-140	6.56E-2	4.76E-2	2.92E-2
Ce-141	2.54E-3	1.85E-3	1.13E-3
Pu-239	2.85E-4	2.07E-4	1.27E-4

Table A-10. Reconstructed TRA airborne effluent for the years 1957, 1956, and 1955 (in Ci)

Radionuclide	1957	1956	1955
Gaseous Component			
Ar-41	3.42E+4	8.89E+4	1.81E+5
Kr-85	1.64E+2	4.25E+2	8.66E+2
Kr-85m	1.55E+3	4.02E+3	8.19E+3
Kr-87	5.59E+3	1.45E+4	2.96E+4
Kr-88	5.48E+3	1.42E+4	2.90E+4
Xe-133	5.97E+2	1.55E+3	3.16E+3
Xe-135	5.93E+3	1.54E+4	3.14E+4
Xe-135m	3.27E+3	8.49E+3	1.73E+4
Xe-138	1.83E+4	4.76E+4	9.71E+4
Particulate and Iodine Component			
Rb-88	2.84E+2	7.38E+2	1.50E+3
Rb-89	4.48E+2	1.16E+3	2.37E+3
Cs-138	4.04E+2	1.05E+3	2.14E+3
Ba-139	3.82E+1	9.92E+1	2.02E+2
Na-24	3.66E+0	9.49E+0	1.93E+1
Hg-203	1.43E+0	3.72E+0	7.58E+0
Tc-99m	1.11E+0	2.89E+0	5.90E+0
Y-91m	5.01E-1	1.30E+0	2.65E+0
Co-60	9.62E-2	2.50E-1	5.09E-1
Cr-51	1.27E-2	3.29E-2	6.70E-2
Cs-134	6.86E-2	1.78E-1	3.63E-1
Cs-137	1.70E-1	4.42E-1	9.02E-1
I-131	1.86E-1	4.82E-1	9.83E-1
I-133	1.48E-1	3.84E-1	7.83E-1
Sr-90	7.14E-1	1.85E+0	3.78E+0
Ba-140	1.31E-2	3.41E-2	6.95E-2
La-140	2.15E-2	5.59E-2	1.14E-1
Ce-141	8.35E-4	2.17E-3	4.42E-3
Pu-239	9.36E-5	2.43E-4	4.95E-4

Table A-11. Reconstructed TRA airborne effluent for the years 1954, 1953, and 1952 (in Ci)

Radionuclide	1954	1953	1952
Gaseous Component			
Ar-41	1.22E+5	1.66E+5	7.53E+4
Kr-85	5.82E+2	7.92E+2	3.60E+2
Kr-85m	5.50E+3	7.49E+3	3.40E+3
Kr-87	1.99E+4	2.71E+4	1.23E+4
Kr-88	1.95E+4	2.65E+4	1.21E+4
Xe-133	2.12E+3	2.89E+3	1.31E+3
Xe-135	2.11E+4	2.87E+4	1.30E+4
Xe-135m	1.16E+4	1.58E+4	7.19E+3
Xe-138	6.52E+4	8.88E+4	4.04E+4
Particulate and Iodine Component			
Rb-88	1.01E+3	1.38E+3	6.25E+2
Rb-89	1.59E+3	2.17E+3	9.86E+2
Cs-138	1.44E+3	1.96E+3	8.90E+2
Ba-139	1.36E+2	1.85E+2	8.41E+1
Na-24	1.30E+1	1.77E+1	8.05E+0
Hg-203	5.09E+0	6.93E+0	3.15E+0
Tc-99m	3.96E+0	5.40E+0	2.45E+0
Y-91m	1.78E+0	2.42E+0	1.10E+0
Co-60	3.42E-1	4.66E-1	2.12E-1
Cr-51	4.50E-2	6.13E-2	2.79E-2
Cs-134	2.44E-1	3.32E-1	1.51E-1
Cs-137	6.06E-1	8.25E-1	3.75E-1
I-131	6.61E-1	8.99E-1	4.09E-1
I-133	5.26E-1	7.16E-1	3.26E-1
Sr-90	2.54E+0	3.45E+0	1.57E+0
Ba-140	4.67E-2	6.36E-2	2.89E-2
La-140	7.66E-2	1.04E-1	4.74E-2
Ce-141	2.97E-3	4.04E-3	1.84E-3
Pu-239	3.33E-4	4.53E-4	2.06E-4

made to the system and to operating procedures to reduce the amount of I-131 released.

The equipment was originally designed to contain and control virtually all of the iodine. However, seals were found to be leaking at a centrifuge during certain operations. The design of the seals was changed, additives were placed in the scrubber solution to increase its efficiency, water levels were changed, and pressures were reduced for jetting operations (Cederburg and MacQueen, 1961). A 10,000-ft^3 holding tank was installed that provided time for radioactive decay and for later release under favorable meteorological conditions. However, plant personnel soon realized that not all the iodine was being released in the few hours while the fuel was being dissolved. A set of charcoal beds was installed and first used in August 1958 to remove iodine from the airborne effluent.

A series of three runs was completed in late 1956, before the main RaLa program began, to test the operating procedures. The three runs, numbered differently from the main RaLa runs, dissolved fuel with progressively shorter cooling times of 1138, 27, and 11 days (Legler et al., 1957). Releases from these three runs were included in the 1956 operational release source terms.

The releases from each of the RaLa runs is shown in Table A-12. The data shown are from AEC Health and Safety Division reports (AEC, 1960; AEC, 1961; AEC, 1962; AEC, 1963). The reports also indicate three (Huffman et al., 1963) or four (ERDA, 1977) RaLa runs in 1963. It can be seen that the efforts to control I-131 were effective and that the quantities released declined significantly over the first 2 years. During the time of the RaLa operations, the environmental monitoring program was expanded to assess the impact from the released radionuclides. A description of the program and results is included in the reports referenced above and in Appendix E.

Considerable discussion among the Task Group members preparing this report and the Peer Review Panel members concerned whether to model the RaLa releases as annual or episodic releases. The large number of runs (about 78), the lack of good data on the time history of each release, and reports showing significant quantities of iodine released over periods of several days

Table A-12. Curies released to the atmosphere because of RaLa operations[a]

Run Number	Date of Run	Radionuclide (Ci) I-131	I-132	I-133	I-135	Beta Activity Minus Iodine (Ci)
001	2/1-3/57	230.0	b	b	b	114.6
002	2/20-21/57	351.3	b	b	b	334.7
003	4/5-6/57	80.8	b	b	b	1.3
004	5/19-20/57	42.0	b	b	b	6.6
005	6/24-25/57	158.7	b	b	b	6.8
006	9/11/57	103.4	b	b	b	5.7
007	10/7/57	201.7	b	b	b	14.8
008	10/21/57	234.7	b	b	b	35.7
	Total Activity 1957	1402.6	4030	441	6.65	520.2
009	1/6/58	35.1	b	b	b	27.1
010	2/12/58	154.6	b	b	b	26.6
011	3/13-14/58	55.6	b	b	b	12.6
012	4/16/58	34.9	b	b	b	9.2
013	4/30-5/1/58	176.6	626.3	b	b	22.4
014	5/28/58	111.1	935.0	b	b	28.7
015	6/2/58	205.8	119.0	b	b	53.0
016[c]	8/6/58	27.7	9.9	b	b	14.9
017	8/13/58	75.3	18.7	b	b	77.6
018	10/1/58	70.6	310.6	b	b	62.7
019	10/22/58	45.6	498.8	b	b	30.6
020	10/22-23/58	d	d	d	d	d
021	11/12/58	32.6	120.6	b	b	72.3
	Total Activity 1958	1025.5	3380	484	7.67	437.7
022	2/4/59	104.8	171.8	b	b	16.2
023	2/25/59	14.4	18.8	b	b	16.4
024	3/18/59	40.5	60.0	b	b	8.0
025	4/8/59	4.7	24.8	b	b	11.0
026	4/29/59	3.4	129.9	b	b	27.6
027	5/19/59	2.3	14.3	b	b	21.2
028	6/10/59	2.1	64.9	b	b	27.3
029[e]	7/6-7/59	9.3	79.5	b	b	8.6
030	7/21-22/59	6.4	183.5	b	b	8.5
031	8/11-12/59	2.1	61.4	b	b	18.0
032	9/1-2/59	4.4	29.8	b	b	4.0
033	9/22-23/59	5.0	37.2	b	b	4.6
034	10/13-14/59	10.3[f]	46.2[f]	b	b	30.6
035	11/5-6/59	12.0	112.2	b	b	9.2
036	11/30, 12/1/59	.4	.3	b	b	2.0
037	12/15-16/59	1.4	27.1	b	b	5.9
	Total Activity 1959	223.5	1061.7	99.8	1.51	219.1

Table A-12. (continued)

Run Number	Date of Run	Radionuclide (Ci) I-131	I-132	I-133	I-135	Beta Activity Minus Iodine (Ci)
038	1/26-27/60	9.97	34.3	b	b	8.32
039	2/24-25/60	1.86	11.0	b	b	3.37
040	3/29-30/60	3.95	7.2	b	b	1.41
041	4/19-20/60	1.21	27.2	b	b	1.63
042	6/1-2/60	1.42	23.8	b	b	2.75
043	6/21-22/60	1.36	4.2	b	b	1.39
044	7/12-13/60	3.54	3.6	b	b	3.66
045	8/2-3/60	1.36	11.2	b	b	0.75
046	8/23-24/60	1.09	11.4	b	b	0.42
047	9/13-14/60	.87	9.5	b	b	0.52
048	10/4-5/60	2.79	22.0	b	b	1.00
049	10/24-25/60	2.60	11.0	b	b	26.49
Total Activity 1960		32.02	176.4	11.4	0.135	26.49
050	1/4-5/61	1.5	9.7	b	b	0.7
051	1/17-18/61	18.1	113.6	b	b	4.3[g]
052	2/7-8/61	1.6	15.2	b	b	1.9
053	2/28-3/1/61	1.5	2.4	b	b	0.4
054	3/21-22/61	1.0	13.1	b	b	2.0
055	4/11-12/61	1.0	0.1	b	b	1.0
056	5/2-3/61	1.0	5.0	b	b	0.8
057	5/23-24/61	1.8	24.2	b	b	1.1
058	6/13-14/61	1.6	1.6	b	b	1.5
059	7/10-11/61	2.5	2.8	b	b	4.5
060	8/15-16/61	2.2	1.1	b	b	2.0
061	9/6-7/61	2.7	13.7	b	b	0.9
062	9/26-27/61	2.4	7.4	b	b	1.6
063	10/17-18/61	1.3	6.8	b	b	0.4
064	11/10-11/61	0.4	3.4	b	b	0.2
065, 066	11/28-30/61	1.5	6.5	b	b	45.2
Total Activity 1961		42.1	226.6	14.9	0.177	68.5
067	1/10-11/62	1.1	13.4	b	b	27.9
068	1/30-31/62	2.5	43.3	b	b	10.0
069	2/20-21/62	1.5	13.1	b	b	4.2
070	3/13-14/62	3.2	14.6	b	b	11.5
071	4/24-25/62	2.7	25.6	b	b	6.6
072	8/14-15/62	1.6	7.9	b	b	1.7
073	9/18-19/62	2.4	2.8	b	b	2.2
074	11/27-28/62[h]	11.5	0.9	b	b	0.6
075	12/11-12/62[h]	13.6	8.1	b	b	4.3
Total Activity 1962		40.1	129.7	14.3	0.169	69.0

Table A-12. (continued)

Run Number	Date of Run	Radionuclide (Ci) I-131	I-132	I-133	I-135	Beta Activity Minus Iodine (Ci)
076, 077 078, 079	1/63-2/63	25.0[i]	82.5[b]	8.90[b]	0.105[b]	b

a. Year of Run (Source of Data): 1957-1959 (AEC, 1960); 1960 (AEC, 1961); 1961 (AEC, 1962); 1962 (AEC, 1963); 1963 (ERDA, 1977).

b. Individual values were not reported in the referenced documents. Totals for the year include calculated values.

c. Carbon filter beds first used.

d. No release. Evidently this is an error in numbering.

e. Start of "2-day" runs.

f. Releases from the 1959 ICPP Criticality on October 16 have been subtracted and are included in the episodic release section of this report.

g. Includes 2.4 Ci attributed to the 1961 ICPP Criticality on January 25.

h. Increase of I-131 because of extensive decontamination operations, allowing release through the vessel off-gas system, which is not equipped with liquid and carbon bed scrubbers as is the dissolver off-gas system.

i. Only the total for the year was found in the references.

convinced the Task Group that the releases should be modeled as annual releases. The decision was reexamined after the peer review was completed, and it was not changed. Below are some of the quotations from reports that led the Task Group to conclude that the iodine was not released in short bursts but over a period of days and perhaps weeks.

> Thus about 3,000 curies of total iodine activity were expected to pass through the scrubber within a one-hour period. On this basis, a storage facility was recommended in March, 1956, in order to contain both xenon and iodine

activity. Subsequent AEC approval resulted a 10,000-cubic feet gas holder installation being completed in July, 1957. This was too late to be of use during the RaLa startup program which was considered completed in June, 1957. During the startup program, it became obvious that the gas holder then being installed would not represent the final solution to the RaLa off-gas problem. This was due to iodine being released over a period of days rather then minutes or hours and also due to radioiodine escaping from centrifuges to cell off-gas, which does not pass through the scrubber. The gas holder will contain process off-gas from 10 hours of operation. Thus it will still be useful for holding radioactive xenon until favorable weather conditions allow venting to the atmosphere (Legler et al., 1957, p. 170).

Evolution of iodine-131 did not occur evenly over the discharge periods indicated. A significant fraction is evolved during the run proper and also during sampling and decontamination following each run. Any disturbance (i.e., sparging, sampling, jetting) of iodine-containing solutions tends to liberate iodine-131.

Results from the stack monitor indicated that more iodine-131 is evolved over a period of days following each run than is evolved during the several hours of operation when the run is actually in progress. (Legler et al., 1957, p. 172).

This [the 10,000-cubic foot gas holder] was found to be inadequate since iodine release from the equipment continued at an appreciable rate for several weeks after a run (Stevenson, 1957, p. 59).

The Health Physics Branch has maintained a continuous off-gas monitor on the dissolver off-gas during and after a RaLa run. It has been found that during quiescent conditions in the cell the iodine release will be from five to ten curies a day. Solution transfer or vessel decontamination will raise this to 20 to 50 curies per day.

Iodine-132 is probably responsible for much of the activity several days after processing. This results from decay of tellurium-132 and subsequent release of the iodine when solutions several days old are sparged or jetted (Stevenson, 1957, p. 61).

The 1962 Annual Progress Report (AEC, 1963) states, "Activity listed was released during the period from one RALA [sic] run to, but not including, the next."

A.2.2.4.2 Reconstructing RaLa Releases. Releases to the environment from RaLa operations are summarized in AEC (1960), AEC (1961), AEC (1962), AEC (1963), and ERDA (1977). These reports contain the amount of I-131, I-132 (beginning in April 1958), and gross beta activity minus iodine. Only the I-131 release is known for 1963. Therefore, the quantities of the individual radionuclides that would have been released from the ICPP 76-m stack were reconstructed.

RaLa processed spent MTR fuel elements as soon as they could be removed from the reactor and transported to ICPP. The fission product inventory for an MTR element was calculated using RSAC-4. Anderson et al. (1959) presents some information on the typical MTR elements processed by RaLa. For an element cooled 2 days and 21.6% burnup, the report gives the following inventories for selected radionuclides:

Nuclide	Ci/Element
Sr-89	1.6E+4
Sr-90	115.
Sr-91	2.4E+3
Ba-140	5.5E+4

These inventories were used to establish the power level and irradiation times for MTR elements processed in RaLa.

Strontium-91 was used to establish the power level of the element. Strontium-91 has a short half-life (9.5 hours) and would have reached equilibrium at the end of the irradiation. Using RSAC-4, a 1.61 MW power level for the MTR element is necessary to give the desired 1.6E+4 Ci of Sr-89.

Strontium-90 was used to calculate the total burnup of an element. A 115 Ci Sr-90 inventory in the element would require 35.2 MWd of burnup. At a power level of 1.61 MW, this would require a 21.9 day or a 1.89E+6 second irradiation period.

RSAC-4 was run using the above operating parameters to calculate the inventories of all of the fission products. RSAC-4 calculated an inventory of 1.65E+04 for Sr-89. This is in excellent agreement with the reported

1.6E+4 Ci. RSAC-4 calculated 5.31E+4 for Ba-140. This is reasonably close to the value of 5.5E+4 in the report. It was noted that RSAC-4 calculated 5.55E+4 Ci for La-140, the daughter of Ba-140, agrees very well with the reported Ba-140 value. It is possible that the La-140 was measured using the 1.6 MeV gamma ray and that scientists assumed the Ba-140 was equal to the La-140. Because the Ba-140 and La-140 were only approaching equilibrium, this would have been an easy error to make.

RaLa runs were conducted over a 1-day period until Run 016 on August 7, 1958. After that time, runs were made over a 2-day period. It was assumed that the activity released from RaLa to the ICPP stack occurred linearly over the entire processing time (1 or 2 days). The RSAC-4 runs decayed the activity accordingly. As discussed previously, iodine releases probably occurred over longer periods, allowing more radioactive decay before release.

The iodine release fraction on a normalized RSAC-4 inventory calculation was adjusted to give a final inventory of 1 Ci of I-131. This allowed the relative amounts of the other halogens to be identified. The I-132 was adjusted to be 3.3 times the I-131 value. This is based on the average of measured ratios over 1 year beginning with RaLa Run 013 in April 1958. Measured releases over the subsequent year were used to evaluate the I-132:I-131 ratio of 3.3.

At the time of the RaLa runs, discharges to the stack were measured by bubbling a fraction of the stack effluent through a liquid scrubber. The beta minus iodine activity reported was obtained by evaporating a portion of the liquid sample and gross counting the remainder for beta activity. Therefore, the release fractions for the particulate material in RSAC-4 runs were adjusted to give a total particulate activity equal to that measured at the time of the RaLa runs. This allowed the release of the individual particulate radionuclides that would have been released to the ICPP stack to be calculated.

A.2.2.4.3 Reconstructing ICPP Releases Before 1957. Reprocessing operations began at the ICPP in 1953. As noted above, the RWMIS reports for the years before 1957 contain only the total gross activity released. While

the majority of this released activity was from the noble gases, there also were releases of particulate activity. Therefore, it was necessary to reconstruct the amount of particulate activity released. Releases were reconstructed using the ratio of Sr-90 to Kr-85 measured in 1959. Because the configuration and operation of the ICPP was essentially the same from 1953 to 1959, it was assumed that this ratio remained constant. The ratio of Sr-90 to Kr-85 measured in 1959 is 3.49E-4. During the initial years of operation, insignificant amounts of iodines were released from the ICPP because of the decay times involved before processing.

Two fission product inventories were calculated to represent typical fuels processed at the ICPP. For the years 1953 through 1956, MTR fuel decayed for 1 year was used to calculate releases during normal plant fuel processing. Beginning in 1957, the ICPP began to process zirconium fuels. The typical fuel assumed for 1957 and subsequent years was from the Shippingport Reactor and was decayed 2 years.

During the earlier years of ICPP operation it was common practice to report only the total quantity of fission gases released. During normal operations, this consisted of the sum of Kr-85 and H-3. Based on Shippingport fuel, noble gas fractions of 0.948 for Kr-85 and 0.052 for H-3 were calculated.

The amount of Pu released in the early years of ICPP operation was calculated using the average ratio of Pu to Sr-90 of 2.56E-3 measured in later years.

The estimated composition of ICPP airborne effluent for the radionuclides making a significant contribution to the dose to the public for the years from 1953 through 1963 are shown in Tables A-13 through A-16. The assumed ratio of Pu-238 to Pu-239/240 for unidentified alpha was 6.8 to 1 for 1953 through 1963. Adjustments for decay chain equilibria similar to those described in Section A.2.2.2.1 were made for Ba-140 and La-140 and for Zr-95 and Nb-95 before addition into the source terms in Section A.3.

Table A-13. Reconstructed composition of ICPP airborne effluent for the years 1963, 1962, and 1961 (in Ci)

Radionuclide	1963	1962	1961
H-3	4.52E+03	4.53E+00	8.56E+00
Kr-85m	3.74E+00	8.41E+00	1.59E+01
Kr-85	8.25E+04	8.62E+01	1.63E+02
Kr-87	2.30E-08	5.18E-08	9.79E-08
Kr-88	1.36E-01	3.06E-01	5.78E-01
Sr-89	1.55E+00	2.47E+00	2.44E+00
Sr-90	2.88E+01	1.41E+00	1.73E-02
Sr-91	6.32E-02	1.01E-01	1.00E-01
Y-91	1.73E+00	2.69E+00	2.66E+00
Zr-95	1.81E+00	2.73E+00	2.69E+00
Nb-95	6.58E-01	6.86E-01	6.66E-01
Ru-103	1.23E+00	1.97E+00	1.95E+00
Ru-106	4.64E+00	2.58E-01	3.42E-02
I-129	2.12E-08	3.40E-08	3.56E-08
I-131	2.50E+01	4.01E+01	4.20E+01
Xe-131m	7.80E+02	1.76E+03	3.32E+03
Te-132	2.91E+00	4.67E+00	4.61E+00
I-132	8.25E+01	1.30E+02	2.27E+02
I-133	8.90E+00	1.43E+01	1.49E+01
Xe-133m	5.81E+03	1.31E+04	2.47E+04
Xe-133	2.64E+05	5.94E+05	1.12E+06
Cs-134	4.39E-01	2.22E-02	9.45E-04
I-135	1.05E-01	1.69E-01	1.77E-01
Xe-135m	2.96E+00	6.64E+00	1.25E+01
Xe-135	7.22E+03	1.63E+04	3.07E+04
Cs-136	5.52E-02	8.85E-02	9.27E-02
Cs-137	2.93E+01	1.44E+00	1.76E-02
Ba-140	4.78E+00	7.67E+00	7.57E+00
La-140	5.14E+00	8.24E+00	8.14E+00
Ce-141	2.67E+00	4.29E+00	4.24E+00
Pr-143	4.71E+00	7.57E+00	7.47E+00
Ce-144	4.12E+01	2.56E+00	5.70E-01
Pm-147	1.16E+02	5.64E+00	4.04E-02
Eu-154	5.79E-02	2.84E-03	4.42E-05
Pu-238	3.51E-02	3.75E-03	5.69E-03
Pu-239/240	5.15E-03	5.51E-04	8.37E-04

Table A-14. Reconstructed composition of ICPP airborne effluent for the years 1960, 1959, and 1958 (in Ci)

Radionuclide	1960	1959	1958
H-3	6.05E+00	6.41E+03	7.81E+03
Kr-85m	1.12E+01	2.12E+01	2.19E+01
Kr-85	1.15E+02	1.17E+05	1.42E+05
Kr-87	6.91E-08	1.32E-07	1.38E-07
Kr-88	4.08E-01	7.81E-01	8.14E-01
Sr-89	9.38E-01	7.56E+00	1.48E+01
Sr-90	4.31E-01	4.09E+01	4.98E+01
Sr-91	3.85E-02	4.31E-01	1.02E+00
Y-91	1.02E+00	8.29E+00	1.61E+01
Zr-95	1.03E+00	8.47E+00	1.64E+01
Nb-95	2.60E-01	2.35E+00	4.29E+00
Ru-103	7.49E-01	6.05E+00	1.18E+01
Ru-106	8.11E-02	6.65E+00	8.18E+00
I-129	2.71E-08	1.84E-07	8.35E-07
I-131	3.20E+01	2.24E+02	1.03E+03
Xe-131m	2.34E+03	3.17E+03	2.41E+03
Te-132	1.77E+00	1.51E+01	3.06E+01
I-132	1.76E+02	1.55E+03	3.38E+03
I-133	1.14E+01	9.98E+01	4.84E+02
Xe-133m	1.74E+04	2.49E+04	2.02E+04
Xe-133	7.92E+05	1.09E+06	8.43E+05
Cs-134	6.81E-03	6.24E-01	7.62E-01
I-135	1.35E-01	1.51E+00	7.67E+00
Xe-135m	8.84E+00	1.71E+01	1.89E+01
Xe-135	2.17E+04	3.81E+04	3.73E+04
Cs-136	7.06E-02	4.97E-01	2.23E+00
Cs-137	4.39E-01	4.17E+01	5.07E+01
Ba-140	2.91E+00	2.37E+01	4.67E+01
La-140	3.13E+00	2.53E+01	4.96E+01
Ce-141	1.63E+00	1.31E+01	2.57E+01
Pr-143	2.87E+00	2.32E+01	4.56E+01
Ce-144	8.21E-01	5.97E+01	7.39E+01
Pm-147	1.72E+00	1.64E+02	2.00E+02
Eu-154	8.70E-04	8.21E-02	1.00E-01
Pu-238	3.12E-02	9.07E-02	1.12E-01
Pu-239/240	4.55E-03	1.33E-02	1.64E-02

Table A-15. Reconstructed composition of ICPP airborne effluent for the years 1957, 1956, and 1955 (in Ci)

Radionuclide	1957	1956	1955
H-3	5.51E+03	2.81E+03	2.71E+03
Kr-85m	7.29E+00	5.52E-15	
Kr-85	1.00E+05	5.40E+04	4.95E+04
Kr-87	4.61E-08		
Kr-88	2.71E-01		
Sr-89	1.92E+01	2.07E+01	1.73E+01
Sr-90	3.50E+01	1.99E+01	1.73E+01
Sr-91	9.19E-01	4.71E-09	
Y-91	2.11E+01	4.36E+01	3.72E+01
Zr-95	2.13E+01	6.36E+01	5.45E+01
Nb-95	5.72E+00	1.34E+02	1.16E+02
Ru-103	1.53E+01	4.46E+00	3.38E+00
Ru-106	5.87E+00	2.03E+01	1.77E+01
I-129	1.29E-06	3.24E-08	9.45E-17
I-131	1.40E+03	1.11E+01	
Xe-131m	1.65E+03	3.22E+02	
Te-132	3.41E+01	1.50E-01	
I-132	4.03E+03	9.83E+00	
I-133	4.41E+02	6.73E-03	
Xe-133m	9.16E+03	1.16E+02	
Xe-133	4.62E+05	2.54E+04	
Cs-134	5.38E-01	7.97E-01	6.93E-01
I-135	6.65E+00	2.08E-11	
Xe-135m	6.97E+00	2.17E-10	
Xe-135	1.24E+04	2.76E-04	
Cs-136	3.17E+00	3.69E-02	
Cs-137	3.57E+01	2.02E+01	1.76E+01
Ba-140	5.85E+01	1.31E+00	2.31E-05
La-140	6.31E+01	1.50E+00	2.65E-05
Ce-141	3.32E+01	3.37E+00	1.91E+00
Pr-143	5.79E+01	1.40E+00	7.23E-05
Ce-144	5.40E+01	2.77E+02	2.41E+02
Pm-147	1.40E+02	6.69E+01	5.82E+01
Eu-154	7.05E-02	4.81E-02	
Eu-156			1.17E-06
Pu-238	7.78E-02	4.44E-02	3.87E-02
Pu-239/240	1.15E-02	6.53E-03	5.69E-03

Table A-16. Reconstructed composition of ICPP airborne effluent for the years 1954 and 1953 (in Ci)

Radionuclide	1954	1953
H-3	1.93E+03	7.75E+02
Kr-85	3.52E+04	1.41E+04
Sr-89	1.23E+01	4.92E+00
Sr-90	1.23E+01	4.92E+00
Y-91	2.65E+01	1.06E+01
Zr-95	3.87E+01	1.55E+01
Nb-95	8.27E+01	3.31E+01
Ru-103	2.40E+00	9.61E-01
Ru-106	1.26E+01	5.03E+00
I-129	6.72E-17	2.69E-17
Cs-134	4.93E-01	1.97E-01
Cs-137	1.25E+01	4.99E+00
Ba-140	1.64E-05	6.55E-06
La-140	1.89E-05	7.54E-06
Ce-141	1.36E+00	5.44E-01
Pr-143	5.14E-05	2.06E-05
Ce-144	1.72E+02	6.86E+01
Pm-147	4.14E+01	1.65E+01
Eu-154		1.19E-02
Eu-156	8.31E-07	
Pu-238	2.75E-02	1.10E-02
Pu-239/240	4.04E-03	1.62E-03

A.2.2.5 1960 Through 1966 ANL-W Releases

The ANL-W releases were based on available discharge records in the RWMIS data base, original waste reports, and discussions with ANL-W personnel. The Experimental Breeder Reactor No. 1 (EBR-I) achieved initial criticality in 1951. Effluent measurements are not available for EBR-I. However, releases are not considered to have been important compared to other Site releases because of the small size of the reactor, the design of the reactor, and the present low effluents from the EBR-II reactor. The first year of operation at the present location was 1960; the ANL-W airborne effluents for 1966 through 1960 are shown in Table A-17.

Table A-17. Composition of ANL-W airborne effluents for the years 1966 through 1960 (in Ci)

Radionuclide	1966	1965	1964	1963	1962	1961	1960
H-3	3.49E+0						
Ar-41	1.35E+2	1.32E+2	1.24E+2	2.06E+2	1.87E+2	1.06E+2	1.77E+2
Kr-85	1.58E+3	5.39E+2					
Ru-106		3.40E-1					
I-131		1.40E+0					
Xe-133	2.53E+3	1.82E+4	1.50E-4				
Xe-135			7.81E+3				

A.2.3 Episodic Releases

The second major category of releases at the INEL is short-term or episodic releases to the air. Included are significant releases from short-term operations, experiments, tests, and other events that must be treated as distinct and not averaged over the year. Most of these releases occurred over a period of less than 1 hour. Some releases in the Initial Engine Test (IET) series covered several months. This section presents pertinent information on the dose calculation for these releases, as well as information on the purposes for the tests and safety constraints imposed on the tests. References are noted for further information.

A description of the methods employed to estimate detailed radionuclide release data is given in Section A.2.3.1. Section A.2.3.2 presents background information to aid the reader in understanding the episodic releases. Table A-20 in Section A.2.3.3 outlines the episodic releases described in detail in that section. The release data by radionuclide can be found in Section A.3 and in the microfiche at the back of this volume.

Each of the described tests, reactor incidents, unintentional criticalities, or other episodic releases has been associated with a fission product inventory generated by that particular event. During the 38 years covered by this report, methods of fission product inventory calculation have changed and improved. With time, fission yields, radioactive half-lives, and neutron absorption cross sections have become more accurate, and computer

codes have replaced hand calculations in generating fission product inventories. Also, it has become more important to completely characterize radioactive material releases to the environment. For example, during the LP-FP-2 test at the Loss-of-Fluid Test (LOFT) Facility in July 1985, considerable effort was expended to monitor and document all radioactive material releases from the facility, including all noble gases, iodines, and particulates. In 1956, on the other hand, when IETs #3, #4, and #6 were conducted for the Heat Transfer Reactor Experiment No. 1 (HTRE No. 1) test series, monitoring capabilities were not well developed. For example, noble gases were considered to be relatively unimportant because of their short radiological half-lives; therefore, noble gas releases were not characterized. In most cases where a reactor operation was involved, however, operational characteristics of the reactor were carefully and adequately documented and provide information that can be used to reconstruct the fission product inventory and subsequent release to the environment.

A.2.3.1 Methods to Evaluate Radionuclides Released

Where sufficient information about early episodic releases at the INEL was not available, the fission-product inventory applicable to that release was recalculated using the methods presently employed for generating safety analysis information. This recalculated inventory is the basis for the source term estimates and subsequent offsite dose calculations.

When assigning radionuclides for a release that was not sufficiently characterized in reference documentation, an attempt was made to be thorough in the definition of the released fission-product inventory. If a particular release was questioned, either the quantity of the referenced nuclide(s) was increased or, if the inventory was incompletely characterized, other appropriate nuclides were assigned to the release. The intent was to be complete, thorough, and yet not overly conservative in the estimate of materials released to the environment. Release quantities were decayed for the time estimated for them to reach the INEL Site boundary. The radioactivity released and decayed to the INEL Site boundary is frequently much less than the amount released from the location of the test. This is especially true of the SPERTs and Aircraft Nuclear Propulsion (ANP) tests,

where a large fraction of these releases was radionuclides with short half-lives.

The computer code used for recalculation of fission-product inventories was RSAC-4 (Wenzel, 1990). RSAC-4 calculates an inventory of fission products from the thermal fissioning of U-235. It accounts for radioactive decay of each nuclide and the production of any additional nuclides produced during decay. The code is capable of simulating short transients, steady-state, or cyclic reactor operations. In addition, the code can account for the release fractions of individual elements or groups of elements in the fission product inventory to best simulate the actual airborne release.

The RSAC-4 computer code produces inventories for well over 250 radionuclides. Many of these radionuclides have short half-lives and will decay to negligible quantities during transit to the INEL Site boundary. Others may have longer half-lives, but they contribute little to the total dose received by an offsite individual. To simplify the dose calculation process, a screening procedure was employed to identify those radionuclides that would contribute significantly to the dose from episodic releases.

Two generic types of nuclear operations were evaluated in the screening procedure, representing the fission product spectrum potentially available for release from (a) short, transient nuclear power operation and (b) a sustained, long-term nuclear power operation, followed by a period for decay of the short half-lived radioactivity. Transient operations characteristic of the SPERT-I tests, SNAP 10A Transient (SNAPTRAN) tests, and ICPP Criticalities, generate an inventory of volatile, short half-lived radionuclides with relatively small amounts of long-lived fission products present. Sustained long-term operations, such as operations of the MTR reactor, produce significant quantities of both short- and long-lived radionuclides. By allowing the short-lived fission products to decay, the relative importance of the longer-lived fission products was assessed. The inhalation and air immersion pathways were chosen because they represent relatively simple, straightforward exposure scenarios representing internal and external exposure modes. Radionuclides identified as producing relatively large air immersion doses in

the screening calculations would also be likely to produce relatively large doses if deposited on the ground surface. Radionuclides identified as the primary contributors to the inhalation dose would be likely to include those radionuclides that would also be important to the ingestion dose pathway.

The screening calculations were initially performed in March 1989 during the early stages of the INEL Historical Dose Evaluation. At that time, the RSAC-4 computer code had not been released. An earlier version, RSAC-3 (Wenzel, 1982), was used to calculate inventories of radioactive material present after a transient operation and a sustained nuclear operation. The transient operation was assumed to last for 1.5 ms at a power level of 30,000 MW, with a 2.9 hour transit time to the INEL Site boundary and was similar to SNAPTRAN-2. 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. The sustained operation was characteristic of operations at the MTR reactor. The resulting RSAC-3 inventory for both of the operations was screened using the following methodology:

1. Radionuclides with 0 Ci inventory were deleted.

2. Radionuclides with half-lives equal to less than 10 minutes were deleted.

3. A relative inhalation dose from each radionuclide was calculated, using the formula described in Section C.2.2.4, and parameter values were selected as representative of an episodic release.

4. A relative air immersion dose from each radionuclide was calculated, using the formula presented in Section C.2.2.3, and parameter values were selected as representative of an episodic release.

5. The radionuclides were sorted by relative inhalation dose (i.e., the radionuclide producing the highest inhalation dose was placed

at the top of the list). The remaining nuclides were placed in descending order by dose value.

6. The radionuclides were sorted by relative immersion dose using the procedure described in the previous step.

7. Cumulative inhalation doses and cumulative immersion doses were calculated (i.e., each dose was added to the sum of the preceding doses on the sorted list).

8. Those radionuclides that contributed to 99% of the cumulative *inhalation* dose were selected.

9. Those radionuclides that contributed to 99.9% of the cumulative *immersion* dose were selected.

The list of 47 radionuclides determined to be the primary dose contributors by the screening calculations are shown in Table A-18. Some radionuclides projected to be in the source terms calculated by RSAC-3 did not have dose conversion factors listed in the references DOE (1988a) and DOE (1988b). Additional evaluations were performed to show that these radionuclides are not present in sufficient quantities, nor do they have large enough dose conversion factors to be included in the list of primary dose contributors.

In addition to the list of 47 radionuclides, airborne releases of Ar-41, U-234, U-235, and U-238 were also evaluated for episodic releases. The Ar-41 was produced by neutron activation of stable argon in air and was generally created when cooling air was passed through a nuclear reactor. Uranium-234, U-235, and U-238 were released in some episodes where damage to reactor fuel occurred.

A.2.3.2 Background Information for Episodic Releases

To understand the role that the INEL has played in the development of the U.S. nuclear safety/power program, the reader should know the context

Table A-18. Radionuclides used in episodic release dose calculations

Radionuclide	Half-Life (h)	Radionuclide	Half-Life (h)
Br-84	5.30E-1	I-131	1.93E+2
Kr-85m	4.48E+0	I-132	2.30E+0
Kr-87	1.27E+0	I-133	2.08E+1
Kr-88	2.84E+0	I-134	8.77E-1
Rb-89	2.53E-1	I-135	6.61E+0
Sr-89	1.21E+3	Xe-129m	2.13E+2
Sr-90	2.50E+5	Xe-135	9.08E+0
Sr-91	9.50E+0	Xe-135m	2.56E-1
Sr-92	2.71E+0	Xe-138	2.36E-1
Y-91	1.40E+3	Cs-137	2.65E+5
Y-92	3.54E+0	Cs-138	5.37E-1
Y-93	1.02E+1	Ba-139	1.38E+0
Zr-95	1.54E+3	Ba-140	3.07E+2
Zr-97	1.69E+1	Ba-141	3.00E-1
Nb-96	2.34E+1	Ba-142	1.80E-1
Mo-99	6.60E+1	La-141	3.93E+0
Ru-103	9.44E+2	La-142	1.59E+0
Ru-105	4.44E+0	Ce-141	7.80E+2
Ru-106	8.84E+3	Ce-143	3.30E+1
Sb-129	4.32E+0	Ce-144	6.82E+3
Te-131	4.20E-1	Pr-143	3.25E+2
Te-131m	3.00E+1	Pr-144	2.90E-1
Te-132	7.80E+1		
Te-133m	9.20E-1		
Te-134	7.00E-1		

under which the program was conducted. To provide this understanding, the safety philosophy governing operation at the INEL Site and classification that governed operations at the INEL Site and communication with the public about these operations are briefly discussed.

After the U.S. Navy had completed its nonnuclear gun testing program, the Idaho Testing Station (ITS) was designated as a location where U.S. nuclear concepts or programs could be safely conducted and would not endanger members of the public. By 1950 the spectrum of radionuclides generated during the process of fissioning of nuclear fuel had been well characterized, and it was recognized that uncontrolled release of these fission products to the environment was undesirable. It was also realized that unfavorable situations do arise with the development of a new industry. The ITS was ideally suited

for this nuclear safety program because of its remote location from large population centers. In the event of a catastrophic accident at the Site, distance and meteorological dispersion would reduce released concentrations to values that would not create health or safety problems for members of the public. Also, the ITS was situated in a location where no surface waters flowed from the Site to an offsite location where potential radioactive material could be carried and subsequently ingested by members of the public.

In 1950 the "destructive force of the atom" and the "harmful effects of radiation" were basically understood. An emphasis was placed on conducting INEL facility operations and projects in a manner that would protect the safety of employees and the public. This safety emphasis becomes apparent as the reader peruses the literature describing the results of each of the tests or experiments conducted at the INEL. For each of the projects, experiments, or facilities that have been operated at the INEL Site, a safety analysis or safety assessment was performed for the purpose of identifying the potential hazards that could exist or could arise. These safety analyses were the documents that defined the parameters within which the project or facility could be operated and not pose a hazard to INEL Site employees or to the public. Review and approval of these documents by the contractor and by personnel of the Idaho Operations Office [under the management of the U.S. Atomic Energy Commission (AEC), Energy Research and Development Administration (ERDA), or U.S. Department of Energy (DOE)] had to occur before operations could begin. Under this principle of review and approval, operations have been conducted in accordance with Government regulations on personnel exposure to radiation, waste disposal, transport of hazardous material, construction or operation of facilities, and other regulated activities. Documentation of one of the first safety assessments was the *HTRE Hazards Report* (Gamertsfelder, 1952). This report defined the potential hazards that could arise from operations conducted during the ANP Program, operated by General Electric (GE) for the U.S. Air Force, for testing nuclear engine prototypes. Another safety assessment document prepared for one of the early episodic releases was *Background Information for Nuclear Aircraft Safety Analysis Program*, which defined the hazards that might have arisen during the conduct of the Fission Product Field Release Tests (FPFRTs) conducted at the INEL to support the GE ANP Program (Convair, 1957; Wilks, 1962).

Many of the early projects were classified for national security reasons. One of these projects was the ANP Program. Communication with the public about INEL Site operations in the early years was governed on a "need-to-know" basis. As a result, many aspects of operations at the National Reactor Testing Station (NRTS) during the early years were not released to the public. Some aspects of INEL operations through the years and continuing today are still governed by that principle.

A.2.3.3 Chronological Listing of Episodic Releases

The episodic releases considered in this report are shown in Table A-19. The approximate amount of radioactivity released, in most cases decayed to the INEL Site boundary, is also listed. The total number of curies is shown to indicate the magnitude of the release, but the dose attributed to each release is <u>not</u> proportional to the total number of curies. References for further information are noted. The location of facilities is shown in Figure A-1.

This section contains discussions of the circumstances surrounding over 100 separate releases or test series. Several of the episodic releases have been separated into two or three parts for evaluation. Each discussion contains a brief description of the test, experiment, or incident and reference citations for the information used to determine amounts of radioactivity released. In most cases, extended quotations from the cited material are provided to indicate the type and quality of data available for this analysis. The references are reproduced as they appeared in the report, and obvious errors are indicated with the word [sic].

The discussions below are organized from the most recent to the earliest. The two exceptions are for test series that were conducted over extended periods of time. These are the Controlled Environmental Radioiodine Tests (CERTs) and the ANP Program. The IETs of the ANP Program are more important from a dose perspective than many of the other episodic releases. Therefore, a longer introductory section is included before a discussion of those tests.

Table A-19. Episodic releases

Date	Description	Area (present name)	Total Curies Released	Reference
10/30/88	1988 Ruthenium Release[a]	ICPP	0.17	Hoff et al. (1989)
7/9/85	LOFT LP-FP-2 Test[a,b]	TAN	8800	Hoff et al. (1986), Carboneau et al. (1987)
12/77	CERT Cs Release[a]	NE of NRF	0.1	DOE (1978)
10/17/78	1978 ICPP Criticality[a,c]	ICPP	6200	Casto (1980)
7/8/76	CERT Ce and Cs Release[a]	NE of NRF	0.6	DOE (1978)
6/6/75	CERT Ce and Cs Release[a]	NE of NRF	0.5	DOE (1978)
9/22/72	LDDT-3[a]	ICPP	1000	
1/72	Ruthenium Release[a,d]	ICPP	1	ERDA (1977)
9/71	TSF Evaporator Release[a,d]	TAN	3E-4	ERDA (1977)
8/31/71	LDDT-2[a]	ICPP	4.2	AEC (1972)
3/3/71	LDDT-1[a]	ICPP	4.4	AEC (1972)
10/21/70	CERT K-42 Release[e]	EFS	0.1	AEC (1971)
9/22/70	CERT K-42 Release[e]	EFS	0.3	AEC (1971)
4/24/70	EXCES, Na-5[a]	GRID III	120	AEC (1971)
4/9/70	EXCES, Na-4[a]	GRID III	100	AEC (1971)
4/7/70	EXCES, Na-3[a]	GRID III	6.6	AEC (1971)
4/1/70	EXCES, Na-2[a]	GRID III	75	AEC (1971)
3/25/70	EXCES, Na-1[a]	GRID III	51	AEC (1971)
1/20/70	CERT Kr-85 Lab Release[e]	CFA	2	AEC (1971)
8/69	EXCES, Xe-4[a]	GRID III	300	AEC (1970)
8/69	EXCES, Xe-3[a]	GRID III	300	AEC (1970)
11/5/69	CERT No. 27[a]	EFS	0.5	AEC (1970)
10/1/69	RDT-4[a]	GRID III	5	
9/18/68	CERT S-35 Release[e]	EFS	0.3	AEC (1969)
8/15/68	CERT No. 24[a]	EFS	0.5	AEC (1969)
6/17/68	CERT No. 23[a]	EFS	0.5	AEC (1969)
5/7/68	RDT-3[a]	GRID III	1.8	AEC (1969)
5/7/68	RDT-2[a]	GRID III	1.0	AEC (1969)
5/3/68	EXCES, Xe-2[a]	GRID III	32	AEC (1969)
2/68	Sodium Release and Fire[a,d]	EBR-II	4E-6	ERDA (1977)
11/30/67	RDT-1[a]	GRID III	1.2	
9/22/67	CERT No. 22[f]	EFS	1	Zimbrick and Voilleque (1969)
7/6/67	CERT No. 20[f]	EFS	0.9	Zimbrick and Voilleque (1969)
2/20/67	ETR Fuel Melt Incident[a,g]	TRA	footnote g	Martin (1967), Covington (1969), Francis et al. (no date)
11/7/66	CERT No. 19[a,f]	EFS	0.5	Bunch (1968)
9/8/66	CERT No. 18[a,f]	EFS	0.1	Bunch (1968)
8/24/66	CERT No. 17[a,f]	EFS	0.1	Bunch (1968)
8/24/66	CERT No. 16[a,f]	EFS	0.1	Bunch (1968)
8/5/66	CERT No. 15[a,f]	EFS	0.1	Bunch (1968)
8/5/66	CERT No. 14[a,f]	EFS	0.1	Bunch (1968)
8/3/66	CERT No. 13[a,f]	EFS	0.1	Bunch (1968)
7/26/66	CERT No. 12[a,f]	EFS	1	Bunch (1968)
7/21/66	CERT No. 11[f]	EFS	8	Bunch (1968)
6/14/66	CERT No. 10[f]	GRID III	5	Bunch (1968)
6/7/66	CERT No. 9[a,f]	NE ICPP	0.05	Bunch (1968)

Table A-19. (continued)

Date	Description	Area (present name)	Total Curies Released	Reference
5/31/66	CERT No. 8[a,f]	NE ICPP	0.05	Bunch (1968)
1/11/66	SNAPTRAN-2[h]	IET	2000	Cordes et al. (1967), Berta (1967), Kessler et al. (1967)
11/22/65	CERT No. 7[f]	EFS	1.1	Bunch (1966), Bunch (1968)
9/14/65	CERT No. 6[f]	ICPP	6	Bunch (1966), Bunch (1968)
6/10/65	CERT No. 5[f]	ARA	0.1	Hawley (1966), Bunch (1968)
5/27/65	CERT No. 4[f]	ARA	0.01	Hawley (1966), Bunch (1968)
12/11/64	CERT No. 3[f]	EFS	1	Hawley (1966), Bunch (1968)
9/2/64	CERT No. 2[f]	EFS	1	Hawley (1966), Bunch (1968)
5/10/64	Waste Tank Farm Incident[e]	ICPP	0.07	AEC (1965)
4/14/64	SPERT-I, Test No. 3[h]	SPERT	1.9	Bunch (1965)
4/1/64	SNAPTRAN-3[h]	IET	4800	Berta (1967), Kessler et al. (1967), Cordes et al. (1965), Kessler et al. (1965), Fletcher (1965), Fletcher (1964)
11/10/63	SPERT-I, Test No. 2[h]	SPERT	0.07	Bunch (1965)
5/27/63	CERT No. 1[f]	N of Atomic City	1	Hawley (1964), Bunch (1968)
11/13/62	MTR Fuel Melt Incident[a,g]	TRA	footnote g	Dykes et al. (1965), Smith (1962a), Smith (1962b)
11/5/62	SPERT-I, Test No. 1[h]	PBF	43	Bunch (1965), Miller et al. (1964), Spano (1962), Scott et al. (1963)
12/12/61	ETR Fuel Melt Incident[a,g]	TRA	footnote g	Keller (1962), Gibson et al. (1963)
1/25/61	1961 ICPP Criticality[h]	ICPP	120	Paulus et al. (1961)
1/3/61	SL-1 Accident[h]	ARA	1100	Kunze (1962), GE (1962), Islitzer (1962), Horan and Gamill (1961), Gamill (1961)
3/1-3/30/61	IET #26(B)[h]	IET	3100	Field (1961), Flagella (1962)
12/23/60-2/28/61	IET #26(A)[h]	IET	7000	Field (1961), Flagella (1962)
12/1-12/15/60	IET #25(B)[h]	IET	7800	Highberg et al. (1961)
11/22-11/30/60	IET #25(A)[h]	IET	2400	Highberg et al., (1961)
11/16/60	OMRE Solvent Burning Experiment[e]	OMRE	0.03	AEC (1961)
10/26/60	IET #24 (LIME-II)[h]	IET	4800	Baker (1961), Pincock (1960a)
9/7-10/14/60	IET #23 (FEET #2)[h]	IET	1700	Pincock (1960b), Flagella (1962)
8/25/60	IET #22 (LIME)[h]	IET	4100	Pincock (1960c), GE (1960)
6/29-8/6/60	IET #21 (FEET #1)[g]	IET	2000	Pincock (1960b), Flagella (1962)
5/14-6/10/60	IET #20[h]	IET	7500	Foster et al. (1960), Flagella (1962)
3/1-4/30/60	IET #19(B)[h]	IET	8400	Pincock (1960d), Flagella (1962)

Table A-19. (continued)

Date	Description	Area (present name)	Total Curies Released	Reference
2/17-2/29/60	IET #19(A)[h]	IET	1200	Pincock (1960d), Flagella (1962)
1/6-2/7/60	IET #18[h]	IET	1.4E+4	Highberg (1960)
12/1-12/12/59	IET #17(B)[h]	IET	2200	Evans (1960a), Pincock (1960d), Flagella (1962)
11/2-11/30/59	IET #17(A)[h]	IET	2400	Evans (1960a), Pincock (1960d), Flagella (1962)
10/16/59	1959 ICPP Criticality[h]	ICPP	3.7E+4	Ginkel et al. (1960)
10/9/59	IET #16[h]	IET	300	Showalter (1959), Miller et al. (1960)
7/9-7/11/59	ICPP Plutonium Release	ICPP	0.1	AEC (1960)
6/16-6/24/59	IET #15(B)[h]	IET	1200	Flagella (1962), Evans (1959)
6/3-6/15/59	IET #15(A)[h]	IET	2000	Flagella (1962), Evans (1959)
4/24-5/19/59	IET #14[h]	IET	7500	Pincock (1959)
12/58	Collection Tank Release[a,d]	ICPP	1	ERDA (1977)
11/18/58	IET #13 (HTRE No. 3 Excursion)[h]	IET	940	GE (1959)
10/29-10/30/58	FECF Filter Break[h]	ICPP	3.9	AEC (1960), AEC (1959)
9/26/58	FPFRT-9[h]	GRID III	10	Wehman (1959), Convair (1958)
9/18/58	FPFRT-8[h]	GRID III	100	Wehman (1959), Convair (1958)
9/17/58	FPFRT-7[h]	GRID III	90	Wehman (1959), Convair (1958)
9/4/58	FPFRT-6[h]	GRID III	120	Wehman (1959), Convair (1958)
8/27/58	FPFRT-5[h]	GRID III	140	Wehman (1959), Convair (1958)
8/14/58	FPFRT-4[h]	GRID III	9.6	Wehman (1959), Convair (1958)
8/58	Blower Failure[a,d]	ICPP	1E-5	ERDA (1977)
8/6/58	FPFRT-3[h]	GRID III	9.9	Wehman (1959), Convair (1958)
8/4/58	FPFRT-2[h]	GRID III	9.3	Wehman (1959), Convair (1958)
7/25/58	FPFRT-1[h]	GRID III	10	Wehman (1959), Convair (1958)
7-9/58	Solvent Burner Release[a,d]	ICPP	0.25	ERDA (1977), AEC (1959)
5/2/58	IET #12 (BOOT)[h]	IET	4000	Wilks (1962), Baker et al. (1959)
3/20-4/14/58	IET #11[h]	IET	4200	
3/11-3/27/58	BORAX-IV Test[i]	EBR-I	footnote i	ERDA (1977), AEC (1959)
3/58	Iodine Release[a,d]	ICPP	1	ERDA (1977)
3/1-3/6/58	IET #10(B)[h]	IET	1.4E+5	Flagella (1962), Foster et al. 1958)
12/20/57-2/25/58	IET #10(A)[h]	IET	1.3E+5	Flagella (1962), Foster et al. (1958)

Table A-19. (continued)

7/31-8/28/57	IET #8[h]	IET	1700	Evans (1957a), Flagella (1962)
3/20/57	Fuel Element Burn Test, B[h]	GRID III	74	Brodsky and Beard (1960)
3/20/57	Fuel Element Burn Test, A[h]	GRID III	4.3	Brodsky and Beard (1960)
12/18/56	IET #6[h]	IET	9000	Thornton et al. (1962)
6/29/56	IET #4(C)[h]	IET	1.5E+5	Thornton et al. (1962)
5/24-6/29/56	IET #4(B)[h]	IET	2.1E+5	Thornton et al. (1962)
5/1-5/23/56	IET #4(A)[h]	IET	6800	Thornton et al. (1962)
2/11-2/24/56	IET #3[h]	IET	4.6E+4	Brodsky and Beard (1960), Thornton et al. (1962)
11/29/55	EBR-I Core Meltdown[i]	EBR-I	footnote i	Zinn (1956), Brittan (1958), Stratton (1967)
6/18-7/1/55	NRF S1W Engineering Test[h]	NRF	310	
7/22/54	BORAX-I Excursion[h]	EBR-I	720	Dietrich (1954), Hayes (1956)

a. These episodic releases are included in the operational release source term.

b. The release of radioactive materials was delayed. The total curies shown are those reported in the RWMIS report for LOFT for July and August.

c. Total curies are from RWMIS, October, 1978. See also DOE (1979).

d. Total curies released is taken from ERDA (1977).

e. Not modeled, small release, see text.

f. Total curies released is from CERT reports.

g. The release was included in the operational releases and is not easily separable or substantially different from the operational releases.

h. The activity reported includes the reduction due to radioactive decay that occurred during transport from the point of release to the INEL Site boundary.

i. Sufficient information was not available on these releases to calculate the quantity of radioactive materials released. An examination of the descriptions of the events, reactor power levels, and other similar releases leads the authors of this report to the conclusion that evaluation of the events would not substantially change the annual doses calculated.

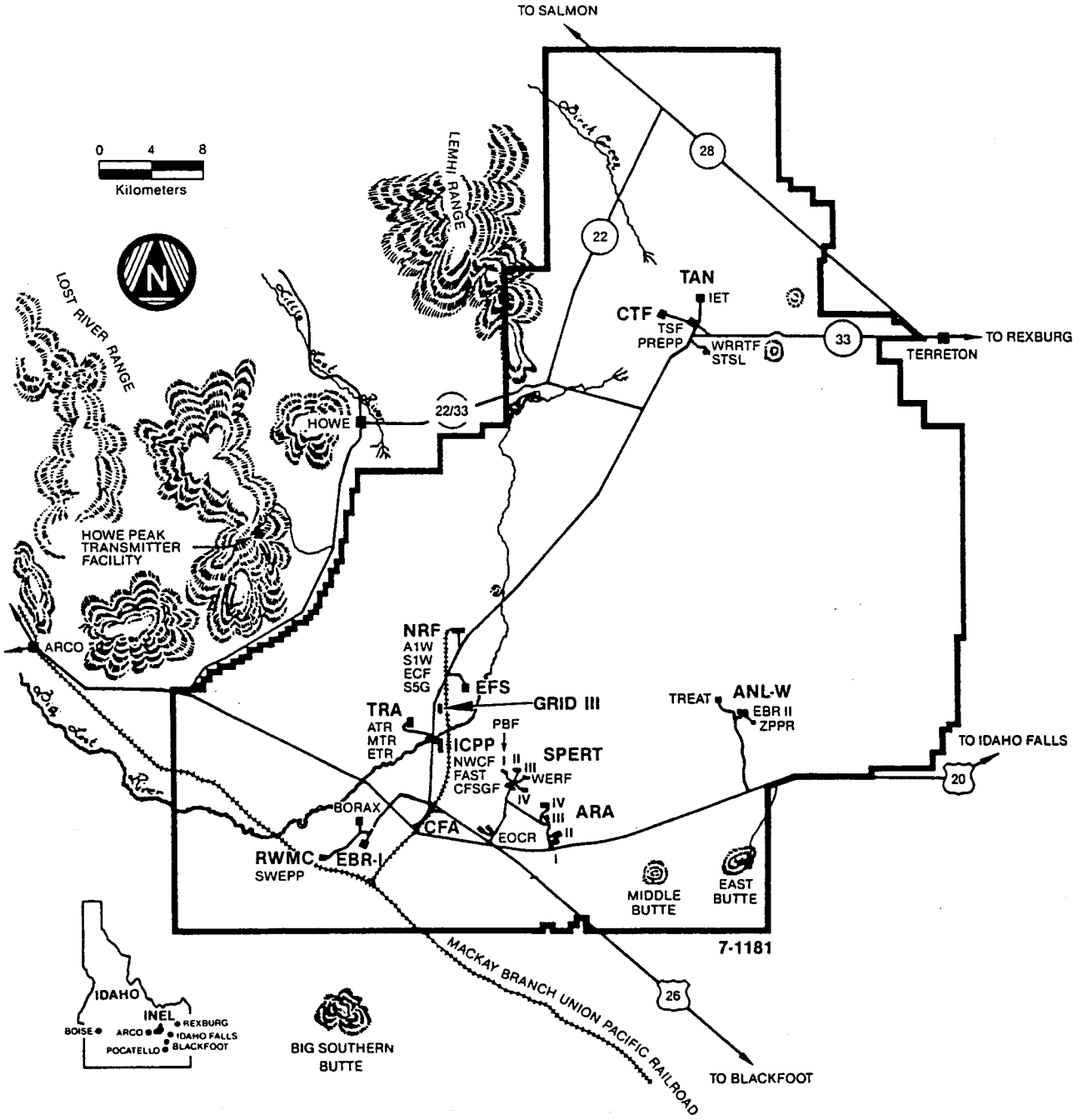


Figure A-1. INEL Site map.

1988 Ruthenium Release (October 30, 1988)

"Approximately 0.17 Ci of Ru-106 was released from the main stack at the ICPP. The calculated exposure to a hypothetical resident at Mud Lake was 0.002 mrem EDE" (Hoff et al., 1989). This small release was included in the operational releases.

LOFT LP-FP-2 Test (July 9, 1985)

This test was the final experiment in a series of eight tests conducted under the support and direction of the Organization for Economic Cooperation and Development, a joint project of 10 nations. The LP-FP-2 Test was designed to simulate a small-break loss-of-coolant accident for the central fuel bundle of the LOFT core, similar to the one experienced at Three Mile Island Unit 2 in March 1979. Based on measurements of radioactivity released during the previous test (LP-FP-1), which involved only minor damage to the fuel cladding, radioactive material was expected to be released during the LP-FP-2 Test from the primary coolant system to the reactor building.

The fission product inventory of the central fuel module was generated for an operation of 430 MWd/t, which was the total burnup on the fuel at the time of the test. The central fuel module consisted of 100 rods of 1000 g of UO_2 each. The small-break scenario resulted in meltdown and slumping of the fuel of the central fuel module and consequent release of fission products to the primary coolant system. The water and the entrained fission products, expelled from the primary coolant system during the blowdown, were captured by the Blowdown Suppression Tank. During the 60-day period following the test, leakage from the fission product monitoring system and the primary coolant system allowed fission products to enter the reactor building. As the decay of this material occurred and it was determined that the release to the environment would not violate LOFT Technical Specifications or Government radiation protection standards, the reactor building ventilation system was operated to evacuate the building air through a filtered and monitored pathway to the environment. Detailed results for this test are documented in Carboneau et al. (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. These quantities are tabulated in Hoff et al. (1986), Table B-10. The release, modeled separately in Hoff et al. (1986), was found to contribute an EDE of 0.003 mrem or 6.3% of the total INEL dose to an offsite individual. In addition, the individual was calculated to reside at Atomic City. For this report, the LOFT LP-FP-2 Test was included in the operational releases. Because of a shorter calculated transit time to the INEL Site boundary, this procedure slightly overestimates the dose to the maximally exposed offsite individual at Atomic City.

1978 ICPP Criticality (October 17, 1978)

At approximately 8:40 p.m. on October 17, 1978, a criticality incident occurred in the first cycle uranium extraction system in the CPP-601 process building. The criticality incident lasted for approximately 20 minutes and caused approximately 3E+18 fissions of U-235. The criticality incident did not result in any significant radiation exposure to personnel. There was no contamination of general plant areas. Releases to the environment were filtered and consisted of only short-lived gaseous activity, their particulate short-lived daughters, and very small amounts of radioactive iodine. Other than trace amounts in one environmental sample, no contamination attributable to the incident was found on or near the ICPP site or in the environment. The release has been included in the operational releases, even though the plume traveled over uninhabited areas to the southwest of the INEL Site (DOE, 1979; Casto, 1980).

1972 Ruthenium Release (January 1972)

Approximately 1 Ci of particulate Ru-106 was released from the ICPP main stack (ERDA, 1977). The release was included in the operational release source term.

Long Distance Diffusion Tests (March 3, 1971 through September 22, 1972)

The Long Distance Diffusion Tests (LDDTs) were jointly conducted by the National Oceanic and Atmospheric Administration (NOAA) and the Health Services Laboratory. The series of controlled releases and concentration measurements evaluated mesoscale atmospheric dispersion of nondepositing tracer gases at the NTRS (now INEL). Known quantities of tracers were released from the ICPP stack at controlled rates under well-documented and continuously monitored meteorological conditions. Air concentrations were measured at distances up to 80 km from the release point. The tracers were radioactive and nonradioactive. The three tests involving radioactive tracers are summarized in Table A-20.

We believe that less than 1000 Ci of Kr-85 were released in the last test. However, that quantity has not been verified. The 3.7 Ci methyl iodide (I-131) release is already included in the RWMIS reports. The remainder of the releases have been added to operational releases.

TSF Evaporator Release (September 1971)

An accidental airborne release occurred from the Technical Services Facility (TSF) liquid waste evaporator. The release was determined to be approximately 266 μCi of Cs-137, 0.0142 μCi of Sr-90, and 0.0142 μCi of Y-90 (ERDA, 1977). The release is included in the operational release source term.

Table A-20. Long Distance Diffusion Test radioactive releases

Date	Test Number	Nuclide	Quantity Released (Ci)	Location
3/3/71	LDDT-1	I-131 methyl	4.4	ICPP stack
8/31/71	LDDT-2	I-131 methyl	4.2	ICPP stack
9/22/72	LDDT-3	I-131 methyl	3.7[a]	ICPP stack
		Kr-85	1000?	

a. Included in the RWMIS reports.

Experimental Cloud Exposure Study (May 3, 1968 to April 24, 1970)

The first series of Experimental Cloud Exposure Study (EXCES) tests in 1968 and 1969 involved the release of Xe-133 over Grid III. The five releases in 1970 were also at Grid III and used Na-24. The objectives of the Xe-133 tests included measuring the total exposures at points 1 m above the ground at several downwind distances using ion chambers, determining the dimensions of the plume, documenting meteorological conditions, and using a balloon support for monitoring equipment. The radioactive releases are summarized in Table A-21.

It is not clear from the reports whether Xe-3 and Xe-4 were two releases of 300 Ci each or one release of 600 Ci. Three releases of approximately 300 Ci each had been planned. No formal reports on the results of the tests were found.

All of the releases in Table A-21 are included in the operational release source terms because of the minor nature of the releases and the lack of specific information on the time of releases.

Relative Diffusion Tests (November 30, 1967 through October 1, 1969)

Four Relative Diffusion Tests (RDTs) were conducted from 1967 through 1969. All four tests were conducted at Grid III. The tests were designed to measure the relative diffusion and deposition of tracers by simultaneous, point-source release of pairs of tracers. In RDT-1, molecular iodine and uranin dye were compared. In RDT-2, uranin dye and methyl iodide were released. Approximately 1½ hours later the same day, molecular iodine and methyl iodine were compared in RDT-3. Information on RDT-4 is incomplete, but the test involved the release of methyl iodide and molecular iodine. The radioactive releases are summarized in Table A-22.

These small releases all originated from the central portion of the INEL Site. Calculations for the CERT test series demonstrates that the offsite dose from tests like these are insignificant. However, the quantities of I-131 released have been added to the operational releases.

Table A-21. Experimental Cloud Exposure Study radioactive releases

Date	Test Number	Nuclide	Quantity Released (Ci)	Location
5/2/68	Xe-1	Xe-133	a	GRID III
5/3/68	Xe-2	Xe-133	32	GRID III
8/69	Xe-3	Xe-133	300	GRID III
8/69	Xe-4	Xe-133	300	GRID III
3/25/70	Na-1	Na-24	51	GRID III
4/1/70	Na-2	Na-24	75	GRID III
4/7/70	Na-3	Na-24	6.6	GRID III
4/9/70	Na-4	Na-24	103	GRID III
4/24/70	Na-5	Na-24	120	GRID III

a. Tests terminated, assumed no substantial release.

Table A-22. Relative Diffusion Test radioactive releases

Date	Test Number	Nuclide	Quantity Released (Ci)	Location
11/30/67	RDT-1	I-131 elemental	1.2	GRID III
5/7/68	RDT-2	I-131 methyl	1.0	GRID III
5/7/68	RDT-3	I-131 methyl	1.0	GRID III
		I-131 elemental	0.84	GRID III
10/1/69	RDT-4	I-131 methyl	5	GRID III
		I-131 elemental	1	

Sodium Release and Fire (February 1968)

Approximately 80 gal of sodium were accidentally released and immediately ignited in the EBR-II Sodium Boiler Plant building control room. The sodium contained approximately 4 mCi of radioactive Na-24. Approximately 0.4 μCi was subsequently released to the atmosphere (ERDA, 1977). The release was insignificant and was not modeled separately.

ETR Fuel Melt Incident (February 20, 1967)

This incident occurred as the result of a piece of tuck tape being inadvertently left in the reactor vessel during the previous shutdown. The piece of tape caused only portions of one fuel element, E-018-D, to be damaged as a result of coolant channel blockage. Melting occurred in Plates 5 through 13, and the severity of the damage varied from 1 x 2-in. to 2 x 18-in. melt zones. Approximately 7.398 g of fuel were lost (Martin, 1967; Covington, 1969; Francis, no date). Airborne effluent releases from this incident were monitored by the ETR stack monitor and are included under operational releases.

SNAPTRAN-2 (January 11, 1966)

SNAPTRAN-2 was the second of two SNAPTRAN tests and was described in Cordes (1967, p. 1):

> As a continuation of the safety program inaugurated by the United States Atomic Energy Commission to assess the nuclear and radiation safety problems of small space reactors, a second SNAP 10A/2 reactor was intentionally destroyed in a Reactor Safety Test Program conducted by Phillips Petroleum Company at the National Reactor Testing Station in Idaho.
>
> As part of a series of safety tests, designated the SNAPTRAN Program, test versions of two SNAP 10A/2 reactors were subjected to severe reactivity insertions which resulted in their complete destruction. The first of these two tests SNAPTRAN-3, was designed to provide information on the radiological consequences of the accidental immersion of a SNAP 10A/2 reactor in water or wet earth such as

could occur during assembly, transport, or a launch abort. The second of these tests, SNAPTRAN-2, was designed to provide information on the dynamic response, the fuel behavior, and the inherent shutdown mechanisms of these reactors in an open air environment (Cordes, 1967, p. 1).

The SNAPTRAN tests were conducted with the reactors and support equipment mounted on a special four-rail flatcar so that the reactor could be moved back into the TAN Hot Shop for inspection and investigation. The configuration and content of uranium and beryllium for the SNAPTRAN-2 test was the same as for the SNAPTRAN-3 test (of April 1, 1964), and the reactivity insertion was effected by the simultaneous 180 degree rotation of four diametrically opposed beryllium control drum assemblies. The operational history of the reactor creating the major portion of the fission product inventory is quoted from the same reference: "During the 1 msec prior to and the 0.5 msec following peak power, a total of 54 MW-sec of nuclear energy was released" (Cordes, 1967, p. 15). The test was conducted at 9:51 a.m. on January 11, 1966, with extensive and elaborate radiological and meteorological support, surveillance, and control capabilities (Cordes, 1967, pp. 4-11). The meteorological constraints for the test were:

- Wind directions from the S-SW (180 degrees-240 degrees) to ensure that the cloud pass over the heavily instrumented section of the grid.
- Wind speed between 3 and 18 m/s The upper limit of 18 mps was established to assure safe aerial monitoring, and the lower range restriction was added to obtain a predictable stable wind which was not subject to rapid directional changes.
- No imminent precipitation in the vicinity of the test. This restriction avoided the possibility of washout of radioactive material before the cloud could be sampled.
- Vertical temperature profile of neutral to light lapse conditions required to prevail for 30 minutes following the test to assure adequate diffusion of radioactive debris.

To prepare for the test, ARFRO conducted a careful weather analysis of pressure systems and trends for several weeks prior to the test. On January 8, 1965 [sic], ARFRO predicted that favorable weather conditions would occur on several days beginning

January 11, 1966, and test preparation was initiated (Cordes, 1967, p. 11).

The instrumented aircraft began monitoring the cloud radiation immediately and followed the cloud on a downwind course determined from visual observations of a balloon released by ARFRO. The plane, while flying at 150 meters above the terrain, intercepted the cloud 300 meters downwind from the reactor. At this time a detector aboard the craft recorded a dose rate of 500 mR/hr. At this corresponding downwind distance a ground level dose rate of 2.7 R/hr was recorded by a remote area detector. The aircraft and ground level telemetering stations continued to monitor the cloud as it moved beyond the site boundary (10 kilometers downwind). The cloud was followed out to a distance of about 30 kilometers at which point radiation levels were not distinguishable from background radiation, which is about 0.2 mR/hr (Cordes, 1967, p. 12).

The SNAPTRAN-2 test confirmed the results of the SNAPTRAN-3 test that a reactivity accident with a "virgin fueled" SNAP 10A/2 reactor does not pose any undue hazard to the general public. The total integrated radiation exposure dose at the NRTS site boundary (10^4 meters) was less than 10 mR for both tests. Likewise, the spread of contamination was limited to a radius of 200 meters from the reactor following both tests (Cordes, 1967, pp. 43-44).

The fission product inventory used for this SNAPTRAN-2 analysis was generated with the RSAC-4 computer code by programming it for the operation described above and the release fractions as described in Cordes (1967), Table VIII, p. 26, which gives the percentage of fission products released to the atmosphere as 75% of the noble gases, 70% of the iodines, 45% of the tellurium, 4% of the solids, and 21% of the total fission product inventory. The uranium airborne release for this test was assumed to amount to 4% of the 4.75 kg of uranium present in the core. The argon-41 generation (4780 Ci) was conservatively assumed to result from the neutrons that escaped from the core.

ICPP Waste Tank Farm Incident (May 10, 1964)

The incident occurred when a steam-flushing operation was underway to remove radioactive contamination from three pipelines to permit their tie-in to new lines. During this operation, a leak developed in a hose coupling. Contaminated fluid issued from this leak, along with steam, and was rapidly dispersed by a high wind over an area of approximately three acres inside the plant fence. Contamination above background was found over an area of approximately ten acres outside the plant fence (AEC, 1965).

Because of the moisture present, large deposition rates occurred near the release point. An estimate of the amount of activity that could have remained airborne has been made based on a direct radiation measurement of approximately 0.1 mrad/h (3 ft above the ground level using an open window instrument) at a downwind distance of approximately 500 m. The open window instrument reading is assumed to equate to the dose expected to the skin.

The activity that remained airborne was estimated to be about 70 mCi. The radionuclides releases were similar to those from the FECF Filter Break of October 30, 1958, but the quantities were about 2% of those released in the FECF Filter Break. The release was not modeled because of the low quantity of radioactivity released, the good dispersive characteristics at the time of the release, and the long distance to the INEL Site boundary.

SPERT-I, Test No. 3 (April 14, 1964)

The SPERT-I testing program consisted of three series of tests, each series was concluded with a short-period excursion during which fuel melting or core damage was anticipated to occur. Because the SPERT-I, Test No. 2 did not produce widespread cladding failure, a third test, SPERT-I, Test No. 3, was planned to determine if more severe damage would occur at higher fuel temperatures (Bunch, 1965, p. 38).

The SPERT-I, Test No. 3 was initiated at 1:14 p.m. on April 14, 1964. The excursion lasted approximately 1.55 ms and had an energy release of 165 MW-s. Two fuel pins ruptured, but there was no significant mechanical damage to the core or to the instrumentation. Radiation fields of 0.2 mrem/h were measured at the SPERT area exclusion fence, 800 m from the reactor building, and the radioactive cloud was tracked to about 4000 m, at which point levels had dropped to background. It was estimated that about 1900 Ci of noble gases were released to the atmosphere. This is about 0.06% of the noble gases produced by a 165 MW-s excursion and about 0.006% of the total fission product inventory. Only Sr-91, Y-91, Sr-92, Y-92, and Ba-139 (radioactive daughter products from the decay of noble gases) were identified by gamma spectrometry (Bunch, 1965).

Prevailing meteorological conditions in the SPERT-I area during the SPERT-I, Test No. 3 were as follows (Bunch, 1965, p. 41):

- Wind speed - 8.5 m/sec at SPERT-I; 7.4 m/sec at the 4800 meter arc
- Wind direction - 245° at SPERT I; 250° at the 4800 meter arc
- Mean wind speed - 8.5 m/sec
- Mean wind direction - 245°
- Stability parameter (n): 0.2
- Horizontal diffusion parameter (C_y) - 0.35 meters$^{(n/2)}$
- Vertical diffusion parameter (C_z) - 0.35 meters$^{(n/2)}$.

The results of air sampling performed during the test were described by Bunch (1965, p. 43):

> The integrated air concentration at N-75 E-75 was about 6.4 μCi-sec/m^3. By correcting this to the time of the excursion, it was estimated that about 8 x 10^{-4} curies were collected from the cloud at that point and that the cloud had an average air concentration of about 8 x 10^{-3} Ci/m^3.

The inventory of radioactive fission products in the reactor core was calculated using the RSAC-4 computer code. No earlier operational history was assumed to be present on the core. Because the release was comprised of short-lived noble gases and iodines produced during the excursion, earlier operations would have had a negligible impact on the activity released from this test. Based on the information described above, the release fraction used in the RSAC-4 computer program for estimating the amount of airborne radioactivity released from the SPERT-I, Test No. 3 was 0.0006 for noble gases. Although no halogens were detected, the halogen release fraction was conservatively assumed to be 0.0006, the same fraction determined for noble gases. The damage to the reactor fuel rods as a result of the excursion was very minimal, resulting in the rupture of only two fuel rods. Because of the scrubbing action of the water in the reactor core and the small amount of fuel damage that occurred, the release fraction for fission product solids was

assumed to be insignificant and was set to zero in the RSAC-4 computer code. For the same reasons, it was assumed that there was no airborne release of uranium from this test.

SNAPTRAN-3 (April 1, 1964)

During the first test of the SNAP 2/10A reactor core configuration, the core was rapidly immersed in water to determine the safety of reactors proposed for use to generate power in the space program. The reactor, environmental water tank, and control equipment were mounted on a four rail flatcar so that the reactor could be returned to the TAN Hot Shop for inspection and investigation. The test was conducted at 11:44 a.m. on April 1, 1964, with extensive radiological and meteorological support, surveillance, and controls. The core is described in Cordes (1965, p. 1).

> The SNAPTRAN 2/10A-3 reactor was a modified SNAP 2/10A Flight System reactor. The core was a cylinder approximately 9 inches in diameter by 13 inches long. The fuel rods consisted of an alloy of zirconium-hydride and 10 weight percent of fully enriched uranium, clad with Hastelloy-N. The core was composed of six beryllium inserts and thirty-seven fuel rods, arranged in a close-packed hexagonal array containing 4.75 kg of U-235 and 464 g-moles of hydrogen (Cordes, 1965, p. 1).
>
> The most probable value of the energy release resulting from the destructive power excursion was determined from a number of different measurements to be 45 $\pm$ 4 Mw-seconds (Kessler et al., 1965, p. 3).

Based on the curves of Figures 11 and 12 of Kessler et al. (1965), 1.5 to 2.0 ms is a good average time period over which this nuclear energy was generated.

> Weather played an important role in the destructive test as a specific set of weather conditions had to be met before the test could be initiated. The potential hazards of the test, particularly the release of fission products and beryllium to the atmosphere, required that the test be prohibited until acceptable meteorological conditions prevailed. These conditions, established by the Idaho Operations Office of the AEC in conjunction with the U. S. Weather Bureau, were:

(1) Lapse (unstable) conditions, to persist a minimum of three hours after the test to ensure adequate diffusion conditions

(2) Mean wind direction in the sector between 180 and 240°, with the allowance of an additional 20° on each side of the sector for short term wind fluctuations

(3) Wind speed between 10 and 30 miles per hour, the lower limit to ensure reliability for forecasting a persistent wind direction and the upper limit to assure safe aerial monitoring, and

(4) No precipitation in the area, in order to maintain good sampling conditions and to prevent localized washout of radioactive material (Cordes, 1965, pp. 5-6).

Meteorological conditions at the time of the test are summarized in Table A-23.

Only slight airborne radiological effects resulted from the destructive test. More than 99% of the fission products were retained in the environmental water and reactor fuel remains. The halogens that escaped from the fuel were retained in the water, and, as a result, no airborne iodine was detected. The only fission products detected in the radioactive cloud were noble gases and their daughters. It is estimated that on the order of 3% of the noble fission gases generated during the destructive excursion was released. The radioactivity from the cloud decreased to background at about 21 miles downwind. Sensitive monitors revealed no airborne beryllium contamination in the cloud or in the near vicinity of the reactor.

Direct radiation likewise was moderate. The highest measured gamma dose was 150 rem at a point 12 ft directly above the reactor. This measurement encompassed the direct radiation from the burst and the fission product decay during a one-hour period following the burst. The radiation level at the rim of the environmental tank 2 h after the test was less than 25 rem/h. At 24 h after the test, this had reduced to about 3 rem/h. The peak dose rate from the cloud at 100 meters downwind was approximately 2 rem/h.

As a result of this test, it can be concluded that no significant radiological hazard to the public will result from the rapid water immersion of a SNAP 2/10A reactor. In addition, the information gained from the test will assist in predicting the behavior of similarly fueled reactors under accident conditions (Cordes, 1965, p. iv).

Table A-23. SNAPTRAN-3 meteorological conditions[a]

Instrument	Direction (degrees)	Speed (mph)
IET 20 feet	203	22.2
IET 200 feet	208	25.5
Station 179	209	17.7
Station Monteview	197	18.6
Tetroon (released after test)	209	27.9

a. Cordes (1965)

The extensive and elaborate methods and grid system established for monitoring the radiological and meteorological conditions following the test are described in Cordes (1965, pp. 3-18).

> The monitoring airplane intercepted the cloud at approximately one mile from the test site and followed it for twenty-one miles before cloud dispersion and radioactive decay reduced the radiation levels to background. At eleven miles from the test site, the airplane measured a peak reading of 0.1 mrem/hr while flying at approximately 1200 feet above the terrain (Cordes, 1965, p. 21).
>
> Although much of the grid instrumentation was oriented towards highly sensitive detection of airborne radioactive iodine, iodine was not detected in the radioactive cloud. Several days following the test, trace quantities of iodine were detected, evolving from the environmental tank and from the vent of the underground waste tank into which the environmental tank had been drained. Since the samplers detecting these trace levels of iodine were the same as those used on the grid (one-inch-deep beds of activated charcoal), it is concluded that trace quantities of iodine in the cloud also would have been detected.
>
> The minimum concentration of iodine detectable by the sampling and counting system under test conditions was 8.7×10^{-9} μCi/cc. This assumes 100% collection efficiency for the charcoal bed, a cloud sampling time of 10 s, and a sampling rate of 20 cfm. Using the upper limit cloud volume calculated earlier (1.45×10^{6}), this concentration yields an upper limit of 3.6×10^{-4} Ci of iodine that could have been released without detection. The results of the field measurements and the sensitivity calculations, therefore, indicate that the iodine release from the destructive excursion, if any, was insignificant (Cordes, 1965, pp. 39-40).

> Only daughters of noble fission gases (3-sec Kr-92, 10-sec Kr-91, 16-sec Xe-140, and 41-sec Xe-139) were collected by air sample filters. In addition, the barium daughter of Xe-139 was also detected on charcoal cartridges in the air samplers, and the cesium daughter of Xe-138 was collected by a fission-gas detector designed to hold the relatively longer lived gases for decay (Cordes, 1965, p. 21).
>
> . . .the analytical techniques discussed in detail in Appendix A determined a release fraction of four percent for Kr-91, three percent for Kr-92, and three percent for Xe-139. One can assume comparable release fractions for other noble gases (Cordes, 1965, p. 42).
>
> The summation of noble gases formed directly by the excursion was calculated by the CURIE code to be 8 x 10^5 curies, about 10 percent of the total fission-product inventory. One percent of the noble gases corresponds to a release of 8000 curies (Cordes, 1965, p. 42).

The fission product inventory used for this SNAPTRAN-3 analysis was generated with the RSAC-4 computer code by programming it for the operation described above (30,000 MW for a period of 1.5 ms). The release was modeled to match the documented release discussed above. In summary, the release fractions assumed were 0.04 noble gases, 0.02 halogens, and 0.0 solids. To be consistent with the release fraction for solids, no uranium was assumed to be released. The Ar-41 release was conservatively assumed to be 4780 Ci, the same as for the SNAPTRAN-2 test of January 11, 1966.

SPERT-I, Test No. 2 (November 10, 1963)

The SPERT-I testing program consisted of three series of tests, each series was concluded with a short-period excursion during which fuel melting or core damage was anticipated to occur. The primary objective of the second series of tests was to study the destructive effects produced by severe power excursions in a low-enrichment oxide fueled core.

The SPERT-I, Test No. 2 was initiated at 2:47 p.m. on November 10, 1963. The ejection of the transient rod from the core was followed by an excursion, which lasted approximately 2.2 ms and had a total nuclear energy release of 155 MW-s. During the excursion, only 2 of the 590 fuel pins in the reactor

core ruptured, and a pressure pulse of less than 100 psi was observed (Bunch, 1965, p. 30).

The meteorological conditions prevailing in the SPERT-I area during the second test were as follows:

- Wind speed - 6.7 m/sec at SPERT-I; 5.4 m/sec at 4800 meter arc
- Wind direction - 228° at SPERT I; 233° at 4800 meter arc
- Mean wind speed - 6.5 m/sec
- Mean wind direction - 230°
- Stability Parameter (n) - 0.20
- Horizontal diffusion parameter (C_y) - 0.178 meters$^{n/2}$
- Vertical diffusion parameter (C_z): 0.118 meters$^{n/2}$.

The inventory of radioactive fission products in the reactor core was calculated using the RSAC-4 computer code. No earlier operational history was assumed to be present on the core. Because the release was comprised of short-lived noble gases and iodines produced during the excursion, earlier operations would have a negligible impact on the activity released from this test. The results from air monitoring performed during the test were used to estimate the release of fission products from the reactor:

> The crosswind concentrations of the cloud were again used in the analysis of centerline concentrations of the plume axis dilution values, X/Q. The resultant members [sic] were back-corrected to the time of excursion to give an estimate of released fission products. This yielded a calculated release of 530 curies or about 0.002 percent of the 2.4x 10^7 curies produced from a 155 MW-sec excursion (Bunch, 1965, p. 33).

The same reference further states:

> No halogens were detected in this test by gamma spectrometry. Estimates based on the detection limits for I-131 and I-135 indicated that the maximum possible releases for the two isotopes were both less than 0.01 percent.

> Concentrations of airborne radioactivity appeared to be a factor of 100 less than in the first test and the release fraction a factor of 350 less--although the period was about one millisecond shorter and the nuclear energy release about five times as great (Bunch, 1965, p. 37).

The release fractions used in the RSAC-4 computer code for estimating the amount of airborne radioactivity released from the SPERT-I, Test No. 2 were 0.0001 for both noble gases and halogens. (The noble gas release fraction was increased to correspond to the minimum detectable release fraction for halogens.) Because of the scrubbing action of the water in the reactor core and the small amount of fuel damage that occurred, the release fraction for fission product solids was assumed to be insignificant and was set to zero in the RSAC-4 computer code. For the same reasons, it was assumed that there was no airborne release of uranium from this test.

CERT Releases (May 27, 1963 through December 1977)

Controlled Environmental Radioiodine Test (CERT) releases were intentional, planned releases of radioiodine designed for the purpose of determining three basic relationships: (1) the amount of radioiodine in the air to that on the soil and vegetation, (2) the amount of radioiodine on the vegetation to that in the milk, and (3) the quantity in the milk to that in the human thyroid after drinking the milk. The name was changed in 1968 to Controlled Environmental Release Test (CERT) when other radionuclides were released.

The following paragraphs identify the releases and present information related to these releases (Hawley, 1964; Hawley, 1966; Bunch, 1966; Bunch 1968; Zimbrick and Voilleque, 1969).

CERT No. 1

> A total of 970 millicuries of I-131 gas was released at 1500 MST on May 27, 1963, near ground level over a 30-minute period under moderately unstable meteorological conditions and an average wind speed of 6.6 meters per second. About 13% of the total released iodine was deposited on the grid. Thyroid doses to the volunteers averaged 0.39 rad (Hawley, 1964).

> A safety analysis was performed using the conditions specified. These parameters were (a) the release of one curie of iodine-131 under strong lapse conditions (n=0.2) and (b) a moderate wind speed (7 m/sec). This analysis showed that the expected inhalation dose at any NRTS facility which could conceivably be affected would be less than 1 mrad, and that the maximum dose from milk ingestion offsite also would be less than 1 mrad.
>
> Direct radiation doses to those handling the radioiodine were estimated to be not more than 250 mrad/person. Appendix A contains a complete safety analysis of the CERT experiment (Hawley, 1964, pp. 13-14).

CERT No. 2 was conducted on September 2, 1964, when approximately the same quantity of iodine released for CERT I was released. The release for this test occurred at 1:44 p.m. and occurred at the Experimental Dairy Farm (EDF), now Experimental Field Station (EFS).

CERT No. 3 was conducted on December 11, 1964, at 1:54 p.m. The test was a scaled-down version of the first two tests. The test was conducted at the EDF.

CERT No. 4 involved the release of 10 mCi on May 27, 1965, for the measurement of the deposition velocity of elemental iodine (I_2) on grass during stable atmospheric conditions. The test was conducted south of ARA.

CERT No. 5 involved the release of 100 mCi of I-131 on June 10, 1965, at 5:15 a.m., for the same purpose as for CERT No. 4. This test was also conducted south of SL-1.

CERT No. 6 involved the release of 2 to 6 Ci of methyl iodide (CH_3I) from the ICPP main stack for the purpose of determining the behavior of this material in the milk-food chain. The release occurred at 2:00 p.m. on September 14, 1965. The radionuclide shipment of 10 Ci had leaked in transit from the supplier. Some leakage also occurred at the Central Facilities Area (CFA) for several days before the test. The release was modeled as a 6 Ci release, probably covering the total INEL release.

CERT No. 7 involved the release of 1.1 Ci of I-131 at 2:10 p.m. on November 22, 1965, at the EDF for the purpose of determining the behavior of radioiodine in late fall or winter.

CERT No. 8 occurred at 8:40 p.m. on May 31, 1966, and was conducted NE of ICPP. The test involved the release of 0.05 Ci of I-131.

CERT No. 9, which involved the release of 0.05 Ci of I-131, was conducted NE of ICPP at 2:45 a.m. on June 7, 1966.

CERT No. 10 was conducted at 10:40 a.m. on June 14, 1966, at Test Grid-III. This test involved the release of 5 Ci of I-131.

Releases for **CERT No. 11** occurred between 1:23 and 1:31 p.m. on July 21, 1966. This test, involving the release of 8 Ci of I-131 as methyl iodide, was conducted at the EDF for the purpose of evaluating the CH_3I behavior in the milk-food chain.

CERT No. 12. The release of 1 Ci I-131 as methyl iodide occurred at 9:08 p.m. on July 26, 1966, at the EDF.

CERT No. 13. The release of 0.1 Ci of I-131 for this test occurred at 10:20 a.m. on August 3, 1966. The test was conducted at the EDF for the purpose of measuring the deposition of iodine as a function of meteorological conditions.

The release of 0.1 Ci of I-131 for **CERT No. 14** occurred at 2:34 a.m. on August 5, 1966, at the EDF. The purpose of the test was the same as that for CERT No. 13.

The release of 0.1 Ci of I-131 for **CERT No. 15** occurred at 5:40 a.m. on August 5, 1966, at the EDF. The purpose of the test was the same as that for CERT No. 13.

The release of 0.1 Ci of I-131 for **CERT No. 16** occurred at 2:30 a.m. on August 24, 1966, at the EDF. The purpose of the test was the same as that for CERT No. 13.

The release of 0.1 Ci of I-131 for **CERT No. 17** occurred at 5:30 a.m. on August 24, 1966, at the EDF. The purpose of the test was the same as that for CERT No. 13.

The release of 0.1 Ci of I-131 for **CERT No. 18** occurred at 7:55 p.m. on September 8, 1966, at the EDF. The purpose of the test was the same as that for CERT No. 13.

The release of 0.5 Ci of I-131 for **CERT No. 19** occurred at 10:24 a.m. on November 7, 1966, at the EDF. It was conducted for the purpose of evaluating the behavior of radioiodine in late fall.

CERT No. 20 involved the release of 0.90 Ci of I-131 at 2:01 p.m. on July 6, 1967, at the EDF. The test was conducted to test the linear relationship between the wind speed and the deposition velocity at high wind speeds.

During **CERT No. 21** iodine was not released to the environment. It was given to cows in capsule form for the purpose of studying the behavior of iodine in the cow.

The release of I-131 during **CERT No. 22** amounted to 0.52 Ci onto each of two grids. The test was conducted on September 22, 1967, with the release occurring at the EDF between 2:15 and 2:45 p.m. The test was for the purpose of determining the precision of field measurements of V_d by carrying out simultaneous releases of molecular iodine over two grids on which experimental conditions are matched as closely as possible.

CERT No. 23. The release of 0.5 Ci of I-131 was conducted at EDF on June 17, 1968. The cows were started on contaminated green chop the next day. The distribution of iodine as a function of height on the pasture grass was determined. The release was included in the operational releases.

CERT No. 24. The release of elemental I-131, probably 0.5 Ci, was conducted at EDF on August 15, 1968, over Sudan grass. A complete study was made of dairy cattle metabolism of radioiodine deposited on Sudan grass. The release was included in the operational releases.

CERT S-35 Release. The release of 0.32 Ci of S-35 as sulfur dioxide was carried out at EDF on September 18, 1968. The test was performed in cooperation with Dr. A. Clyde Hill of the University of Utah. Because of the small release, the test was not modeled.

CERT No. 25 was conducted in late August or September 1969 to gather additional preliminary data on the effect of Sudan grass on radioiodine metabolism. Dr. B. R. Moss of Montana State University directed this experiment and CERT No. 26. No radionuclides were released as the six cows were "dosed" with radioiodine.

CERT No. 26 was similar to CERT No. 25 with no atmospheric release. The full objectives of this experiment were not met because the Sudan grass froze before CERT No. 26 could be run.

CERT No. 27. The release of 0.5 Ci of Cr-51 was conducted at EDF on November 5, 1968, in a 30-minute period. The release was included in the operational releases.

CERT Kr-85 Lab Release. About 2 Ci of Kr-85 were released during a laboratory CERT experiment from CFA on January 20, 1970. Because of the very small release, compared to the 140,000 Ci of Kr-85 released from the INEL Site in 1970, this test was not evaluated further.

CERT No. 28. This was a test of bovine metabolism performed in cooperation with B. R. Moss and D. J. Hoss of Montana State University. There was no atmospheric release. The test was performed in late July or August 1970.

CERT No. 29. This test was similar to CERT No. 28 and was performed in the same time period.

CERT K-42 Release. About 0.1 Ci of K-42 was released at the EDF on September 22, 1970.

CERT K-42 Release. About 0.3 Ci of K-42 was released at the EDF on October 21, 1970. The K-42 releases are quite small and were not considered further.

CERT Ce and Cs Releases. Three tests were conducted in 1975, 1976, and 1977 to measure and compare the deposition velocities and retention half times of Ce-141 and Cs-134 on sagebrush and squirreltail bottlebrush grass at various seasons of the year. The following tests were conducted at a CERT grid northeast of NRF: (a) June 6, 1975--0.2 Ci of Ce-141 and 0.25 Ci of Cs-134; (b) July 8, 1976--0.3 Ci of Ce-141 and 0.25 Ci of Cs-134; and (c) December 1977--0.1 Ci of Cs-134.

The 1975 and 1976 releases were added to the operational releases. The amount released in 1977 is shown in some reports as 0.2 Ci, but measurements of the aerosol generator following the test indicated that less than half of the original material had been released. That amount, 0.1 Ci, was included in the RWMIS report for 1977.

MTR Fuel Melt Incident (November 13, 1962)

> The event marked the first instance that melting of fuel alloy has occurred in the Materials Testing Reactor core. The minor damage incurred attests to the ability of the core differential pressure monitoring system to cope with a flow blockage situation within individual fuel element positions (Dykes et al., 1965, p. iii).
>
> A piece of dislodged gasket material from the MTR primary coolant system caused a significant flow restriction in at least two flow channels of a standard MTR fuel element. Partial melting of one fuel plate resulted in the loss of approximately 10.5 grams of metal, containing an estimated 0.7 gram of fissionable material.
>
> Contamination of the primary coolant system was not severe, nor were any personnel overexposures received (Dykes et al., 1965, p. ii).

Airborne effluent releases resulting from this fuel melt incident were monitored by the MTR stack exhaust monitor and are included in the operational releases (Dykes et al., 1965; Smith, 1962a; Smith, 1962b).

SPERT-I, Test No. 1 (November 5, 1962)

The SPERT-I testing program consisted of three series of tests. Each series was concluded with a short-period excursion in which fuel melting or core damage was anticipated to occur. The purpose of the first series of tests was to investigate short-period reactor excursions and the factors that might be responsible for the destructive pressure pulses that apparently occurred in the BORAX-I Excursion and the SL-1 Accident.

The SPERT-I, Test No. 1 was initiated at 12:25 p.m. on November 5, 1962, after waiting approximately 2 weeks before meteorological conditions were favorable for the test (Miller et al., 1964). Some of the safety and meteorological constraints on the test are described by Miller et al. (1964, p. 3):

> The test was conducted under special procedures and with specified weather conditions. Safeguards which were adopted included evacuation from the area of all personnel not essential for the test and an alert status with various safety support services at the National Reactor Testing Station. 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 three hour predicted persistence of these conditions after the test. The wind direction was chosen to prevent a possible release of radioactive fissides from being airborne to other NRTS installations.

The actual meteorological conditions in the SPERT-I area are reported in Bunch (1965, p. 17):

- Wind speed - 10.3 m/sec at SPERT-I; 12.9 m/sec at the 4800 meter arc
- Wind direction - 226° at SPERT-I; 238° at the 4800 meter arc.
- Mean wind speed - 11.6 m/sec

- Mean wind direction - 230°
- Stability parameter (n) - 0.25 (neutral)
- Horizontal diffusion parameter (C_y) - 0.178 meters$^{n/2}$
- Vertical diffusion parameter (C_z) - 0.118 meters$^{n/2}$.

The observed consequences of the test are summarized in Bunch (1965, p. 4):

> The actual excursion had a period of 3.2 msec, a maximum power of 2300 MW, and a nuclear energy release of 30.7 MW-sec. The most surprising feature of this excursion was that about 15 msec after the power peak and long after the expected pressure transient of 40 psig had disappeared, a sudden pressure transient occurred which, it is believed, eventually reached 3600 psig or more and destroyed the core. The mechanism of this delayed pressure pulse is not yet fully understood. Possibly a rapid dispersal of molten fuel was triggered by a steam explosion due to water trapped between fuel plates. It appears that about 3.5 MW-sec of energy was released by the metal-water reaction (with the formation of about 0.4 kg of αAl_2O_3).

Complete fuel plate melting and partial melting were estimated to have occurred in about 8 and 35% of the core, respectively (Miller, 1964, p. 52).

The inventory of radioactive fission products in the reactor core was calculated using the RSAC-4 computer code and included the fission product contribution from previous reactor operations. Considerable effort was made at the time of the test to determine the quantity of radioactive material released. Based on the results obtained from the array of air monitors, the following summary is provided in Miller et al. (1964, p. 47):

> . . . it is estimated that about 2.4 x 10^5 curies were released to the atmosphere as a result of the destructive test. The fisside inventory of the core at the time of the release was almost totally due to the fissioning which took place during the power excursion with only a small contribution from the long-lived radioisotopes from previous tests. Using a value of the nuclear energy release of 31 MW-sec, it is estimated that the total fisside content of the core was 6 x 10^7 curies and that, therefore, this calculation indicates the fraction of this inventory which was released to the atmosphere to be of the order of 0.4 percent.

> Improvements in the calculation of fisside release have been continued by the USAEC-ID Health and Safety Division. These calculations, which are still preliminary, take into account the specific decay schemes of the collected isotopes, Sr-91, Sr-92, and Ba-139 as well as an improved computer calculation of the noble gas inventory of the core at the time of the release. It has been calculated on this basis that about 7 percent of the noble gases were released to the atmosphere. Since neither solid products nor radioiodines were found, it is, therefore, estimated that the fisside release consisted mainly of the noble gases and amounted to about 0.7 percent of the total fisside inventory.
>
> Although radioiodines were not detected following the destructive test, it has been possible to calculate the maximum release of radioiodines which could have taken place undetected. By this approach, it has been established that less than 0.01 percent of the radioiodines were released to the atmosphere.

Additional details concerning the estimated release of halogens during the SPERT-I, Test No. 1 are provided in Bunch (1965, p. 28):

> No halogens were identified by gamma spectrometry; and since they are normally expected to contribute the greatest hazard to the environment, calculations were made to determine what amounts might have been present and not detected. For the grid location N-75 E-75, the maximum activities on one spectrum could have been 1.1×10^2 disintegrations/minute/sample of I-131 and 1.3×10^2 disintegrations/minute for I-135. This amounts to a release of 51 μCi of I-131 and 160 μCi of I-135. Therefore, the maximum possible releases for the given isotopes are estimated to be 0.006 percent for I-131 and 0.0003 percent for I-135. Although the detectable activities on the spectra were essentially the same (110 and 130 disintegrations/minute), calculations involving a higher specific activity and higher fission yield for I-135 created a difference in possible release of two orders of magnitude.
>
> It is apparent, then, that little or no iodine was released. Although water was spewed from the reactor and gross damage occurred, over 99 percent of the fission product inventory was retained in or near the reactor vessel. Of the fraction that was released, the activity had decayed by a factor of 10^5 one day after the excursion.

The summary statements indicate some uncertainty as to the fraction of noble gases and halogens present in the reactor that were released during the test. Therefore, the release fractions used in the RSAC-4 computer code were assumed to be 0.1 for noble gases, 0.0001 for the halogens, and 0.00001 for

the solids. Uranium in the reactor fuel was 93% enriched and was assumed to be released in the same fraction as the fission product solids (0.00001). These release fractions should provide a bounding estimate of the airborne radioactivity released from this test.

ETR Fuel Melt Incident (December 12, 1961)

This incident was caused by a plexiglas sight glass that was inadvertently left in the reactor vessel during the previous shutdown. During initial stages of coolant circulation, the sight glass broke and eventually caused flow blockage of several fuel elements, causing portions of these elements to melt. Through the coolant degassification system, there was an increase of principally noble gases released to the atmosphere. The release was continually monitored by the ETR stack monitor. After the incident had been noticed because of increased radiation from the primary coolant, the reactor was shutdown, the damaged fuel elements were removed, and the system was cleaned of tramp fuel that had been released from the damaged fuel elements. However, because some fuel remained to circulate with the coolant and the fission products released from this tramp fuel as it passed through the core region, an increased amount of activity was released to the atmosphere (Keller, 1962; Gibson et al., 1963). Airborne effluent releases from this fuel melt incident were monitored by the ETR stack exhaust monitor and are included under operational releases.

1961 ICPP Criticality (January 25, 1961)

At approximately 9:50 a.m. on January 25, 1961, a criticality incident occurred at ICPP in vessel H-110 (Paulus et al., 1961). The magnitude of the criticality incident was estimated to be 6E+17 fissions.

> A nuclear incident involving an enriched uranium solution occurred in a first cycle product evaporator at the Idaho Chemical Processing Plant, National Reactor Testing Station, at approximately 0950 on January 25, 1961. The reaction resulted from an accidental air lifting of a quantity of solution from a geometrically safe region of the evaporator into the 24-inch diameter vapor disengaging chamber. An energy release of 20 megawatt-seconds (6×10^{17} fissions) apparently occurred as a single burst. No significant radiation exposure to personnel,

> contamination of facilities or environs, property damage, or product loss resulted. Details of events leading to the incident, recommendations and corrective measures are reported (Paulus et al., 1961, p. iii).

Because no reports or letters could be found that indicated the quantity of radioactivity released to the environment, it has been necessary to reconstruct the releases for the 1961 ICPP Criticality. Experience from the other two criticalities at ICPP (1959 and 1978) indicates that releases to the environment from criticality incidents occurring in process vessels consist primarily of iodines, noble gases, and the short-lived daughters of the noble gases.

It has been conservatively assumed that 100% of the noble gases produced during the 1961 ICPP Criticality were released to the off-gas ducting to the stack. No credit was taken for the plateout of the short-lived daughters in the off-gas ducting to the stack. Releases of the iodines were assumed to be 25% of the total produced during the criticality. This is approximately twice the highest release fraction measured during the other two criticalities at ICPP.

The radioactivity released to the environment from the 1961 ICPP Criticality has been calculated using the RSAC-4 computer code. Activities reported have been decayed 2.3 hours, the time calculated for the plume to reach the INEL Site boundary.

SL-1 Accident (January 3, 1961)

The Stationary Low Power No. 1 Reactor (SL-1) was the smallest known power reactor when it began critical operations in August 1958. This direct cycle, natural circulation, boiling water reactor was part of the Army program to develop simple and compact package power plants to be transported by air to remote Arctic sites. The SL-1 reactor was shut down on December 23, 1960, for 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 (Gammill, 1961). The crew was apparently reinstalling the control rod drive mechanisms,

which required lifting the control rods a very short distance to make the connection. It is presumed that while connecting the central control rod, it was lifted far beyond the distance required for connection (ERDA, 1977). At 9:02 p.m., a nuclear excursion and steam explosion occurred in the SL-1 reactor killing two of the personnel and fatally injuring the third member of the crew (Gammill, 1961). At 9:10 p.m. on January 3, 1961, the seven-man fire department crew, responding to a heat sensing alarm for the SL-1 reactor, found radiation fields of 200 mR/h at the gate to the area 200 ft from the reactor building (Horan and Gammill, 1961).

Fuel for the reactor was a U-Al alloy clad with Al formed into plates similar to an MTR type fuel assembly (Kunze, 1962). U-235 enrichment was 93% and final uranium content of the fabricated plates was about 8 wt % with some variation from plate to plate. Analysis of debris from the bottom of the reactor vessel containing 1177.3 g of uranium shows that 360.2 Ci of Cs-137, 1433.9 Ci of Ce-144, 294.5 Ci of Sr-90, 29.2 Ci of Zr-Nb-95, 28.4 Ci of Cs-134, and 159 Ci of Ru-106 were contained in the debris. This analysis, performed at ICPP, stated that the total core, based on 930 MWd of operation should have contained 3050 Ci of Cs-137, 15000 Ci of Ce-144, 3070 Ci of Sr-90, and 400 Ci of Ru-106. Operational history of the core is given on page 38 of Kunze (1962).

> The supplementary flux wire data reported in Appendix E indicates that the energy release was 80 Mw-sec in the center 16 elements (78 Mw-sec reported in IDO-19311), and 53 Mw-sec (50 Mw-sec previously) in the outer 24 elements, for a total nuclear energy release of 133 Mw-sec. This distructive [sic] nuclear energy release occurred after about a seven day shutdown at the end of an operation that generated 931 MWD. The estimated standard deviation of this value is approximately ±10 MW-sec, assuming spectral changes during the excursion were not such that a significant fraction of the fissions occurred in the resonance region (Kunze, 1962, p. 97).
>
> The release of fission products to the reactor operating room and to the outside of the building was affected by the containment provided by the pressure vessel and building at the time of the accident. The release of fission products from the fuel was dominated by the disintegration or complete melting of a portion of the reactor (about 20%) and the release of virtually all the fission products from these regions.. . . Less that 1/2% of the iodine-131 and a negligible fraction of the nonvolatile fission product inventory was found on the desert (Kunze, 1962, p. 99).

> One of the first actions taken following notification of a radiological incident was the activation of an 11 station, high volume air sampling network by means of a telephone signal. These sampling units consisted of a Staplex air sampler fitted with an MSA-2133 prefilter for collection of particulates and an activated charcoal filter for the iodine isotopes. One of these stations was located at Atomic City, the nearest off-site population. Gamma spectral analysis of the first sample collected there indicated that the intitial [sic] cloud was essentially all iodine-131. At that location, about 5.3 miles downwind from the SL-1, the average iodine-131 concentration in air was approximately 5×10^{-11} uc/cc [sic] for the first 16 hours following the accident (Gammill, 1961).
>
> Dairy farms are sparse in this section of sagebrush desert and lava rock terrain. A raw milk sampling program was initiated on all 8 farms in the 150 square mile area where iodine-131 contamination of vegetation exceeded 10 times background. Four out of 70 samples indicated 2×10^{-7} uc [sic] of iodine-131 per cc of milk, which is the analytical detection limit. At this time of year, dairy herds are fed in feed lots, thus minimizing the effect of iodine deposition on the range land (Gammill, 1961).

Extensive vegetation sampling, aerial monitoring (starting early January 4), extra radiation monitoring with film badges, extra air sampling, and soil sampling were also conducted.

Air and vegetation sampling indicated that less than 10 Ci of I-131 were released during the first 16 hours and less than 70 Ci over the remaining 30-day period. An air sample from the reactor floor revealed aged fission products plus I-131. From air samples collected at Atomic City during this period (January 3 through February 20, 1961), the calculated infinity thyroid dose was approximately 35 mrad, which is slightly greater than 1% of the annual 3 rad recommended Radiation Protection Guide value for offsite population (Gammill, 1961).

> A total of 18 soil samples were collected from the 3-acre plot composing the project area within the SL-1 perimeter fence. An additional 36 samples were collected radially to a distance of 1/2 mile beyond the fence. Essentially all the radioactive material, with the exception of iodine-131, was contained within the 3 acre plot. Our best estimates indicate the release of approximately 1/10 curie of strontium-90 and 1/2 curie of cesium 137, with appreciably lesser quantities of zirconium-niobium-95, cerium-144, and barium-lanthanum-140 (Gammill, 1961).

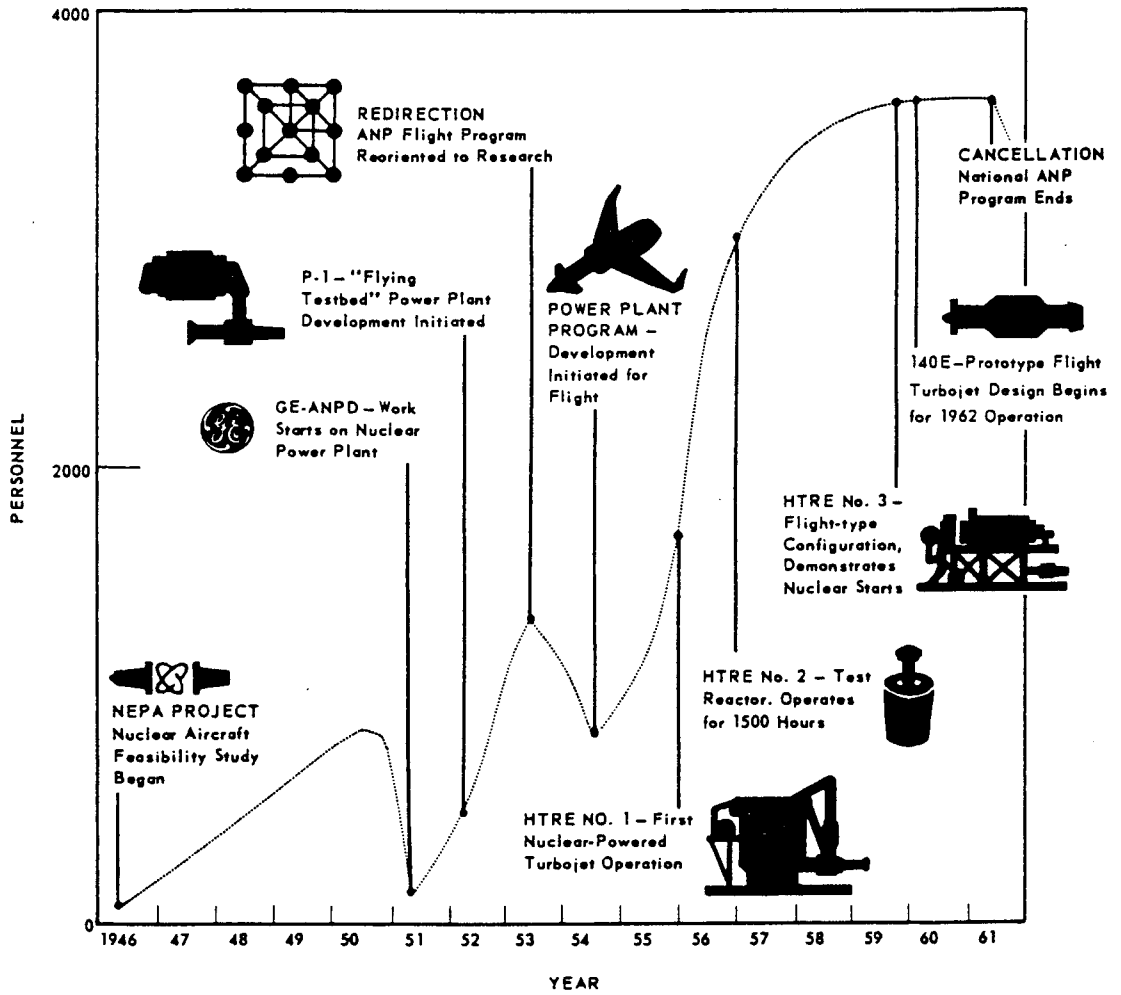
The measured values of I-131, Sr-90, and Cs-137 released to the environment were used as the starting point for the SL-1 supplementary calculations. The measured I-131 air concentration for the first 16 hours after the accident was used to establish the dispersion coefficient. A RSAC-4 calculation of the total SL-1 inventory of radionuclides was compared to the estimated amounts of I-131, Sr-90, and Cs-137 released. The following assumptions were made: (a) all noble gases were released, (b) all iodines were released in proportion to the I-131 release fraction, (c) all strontium isotopes were released in proportion to the Sr-90, and (d) all other particulates were released in proportion to the Cs-137 release fraction. The ingestion pathway for short half-life radionuclides was reduced to 10% of the calculated value because the accident took place in mid-winter when no grass or crops were growing.

Additional references to the SL-1 accident are GE (1962) and Islitzer (1962). Islitzer discusses the role of meteorology in the SL-1 Accident. Detailed dispersion calculations were performed that are in general agreement with the concentrations measured in the environment. However, the measured concentrations of I-131 at Atomic City were higher than the calculated concentrations. The higher measured concentrations were used as the basis for the current dose evaluation.

Aircraft Nuclear Propulsion Program (1953 through 1961)

The Aircraft Nuclear Propulsion (ANP) Program was operated by GE for the U.S. Air Force for testing nuclear engine prototypes. The ANP Program, graphically represented with respect to time in Figure A-2, was supported at the INEL by experiments conducted in the MTR; in the ETR; at GRID III; in the SPERT reactors; in the BORAX I reactor; and at TAN, where the major part of the program was conducted. The ANP experiments that involved significant airborne releases of radioactivity at the INEL were the power operations of the HTRE No. 1, HTRE No. 2, and HTRE No. 3 reactors; the Fuel Element Burn Tests; and the FPFRTs.

The goal of the ANP Program was to investigate the feasibility of developing a nuclear or nuclear/chemical propulsion system for military uses.



A detailed history and chronology is provided in Thornton and Rothstein (1962). Chronology information extracted from: Aircraft Nuclear Propulsion Program hearing before the Subcommittee of Research and Development of the Joint Committee on Atomic Energy, 86th Congress of the United States, First Session, July 23, 1959, United States Government Printing Office, Washington 1959.

Figure A-2. ANP Program with respect to time.

> The power-plant design concept selected for development by the General Electric Company was the direct air cycle turbojet. Air is the only working fluid in this type of system. The reactor receives air from the jet engine compressor, heats it directly, and delivers it to the turbine. The high-temperature air then generates the forward thrust as it exhausts through the engine nozzle. The direct air cycle concept was selected on the basis of studies indicating that it would provide a relatively simple, dependable, and serviceable power plant with high-performance potential (Wilks, 1962, p. 5).

During the 8-year period from 1953 to 1961, three reactor assemblies were constructed that were instrumental in conducting the ANP Program operations. These three HTRE assemblies were designated HTRE No. 1, HTRE No. 2, and HTRE No. 3. These assemblies were used to evaluate reactor control systems, test advanced fuels and moderators, and demonstrate the feasibility of operating a nuclear powered aircraft. The ANP Program ended in 1961 when the program was terminated by President John F. Kennedy.

The HTRE assemblies were mounted on a four-track railroad dolly and were operated at the Core Test Facility (CTF), located at the TAN. The CTF provided the shielded control room; the support utilities required for testing (e.g., water and electrical power); and the instrumented reactor exhaust system. Only one HTRE assembly at a time could be operated at the CTF.

The testing program for the HTRE assemblies was designated as the Initial Engine Tests (IETs). The IETs were numbered sequentially from IET #1 through IET #26. A summary of the IETs performed during the ANP program is provided in Table A-24.

Effluent Monitoring for the Initial Engine Tests

Several types of samplers and monitors were used at the CTF to monitor radioactivity in the airborne effluent from the IET reactor experiments. Measurement and evaluation of the radioactivity in the airborne effluent was an important task beginning with the power testing program for the HTRE No. 1. From an operations standpoint, the monitoring information was used on a real-time basis to help ensure that the reactor was operating normally and that required field monitoring was performed. The continuous stack monitor and

Table A-24. General Electric-Aircraft Nuclear Propulsion Program Initial Engine Test summary

IET Test #/ HTRE No./ Insert Number	IET Test Dates	First Date of Nuclear Operation	Reactor Energy Production (MW-h)	Hours at Test Conditions	Total Test Hours	Brief Test Description	References
1/NA/NA	Unknown	NA	NA	NA	NAV	Engine and Duct Valve Operational Test. The test was not a power operation of the reactor.	Thorton et al. (1962, p. 23)
2/NA/NA	Unknown	NA	NA	NA	NA	Probably the critical test series for HTRE No. 1. The test was not a power operation of the reactor.	Thorton et al. (1962, p. 23)
3/1/NA	12/27/55-2/25/56	1/17/56	349	6.3	40.21[a]	First power operation of the HTRE No. 1 for determining the nuclear characteristics of the engine.	Thorton et al. (1962)
4/1/NA	4/17/56-6/29/56	5/1/56	2065	84	193.94[a]	Second HTRE No. 1 operation to determine suitability of corrections to deficiencies noted in IET #3.	Thorton et al. (1962)
5/1/NA	Unknown	NA	NA	NA	NA	Series of shielding tests. The test was not a power operation of the reactor.	Thorton et al. (1962, p. 23)
6/1/NA	9/24/56-1/3/57	10/12/56	3092	105	257.61[a]	Third HTRE No. 1 operation to investigate basic controls and to conduct endurance testing.	Thorton et al. (1962)
7/2/NA	2/22/57-7/57	NA	NA	NA	NA[b]	Critical experiment and low-power testing of HTRE No. 2 with inserts 1-A and 2-A mockups.	Evans (1957a), Evans (1957b), Flagella (1962), GE (1957)
8/2/1B	7/18/57-8/27/57	7/25/57	~400	33.97	NAV[b,c]	First power operation of the HTRE No. 2. Testing of clad zirconium hydride moderator.	Evans (1957a), Flagella (1962)
9/2/NA	9/57	NA	NA	NA	NA[b]	Critical experiments and low-power testing of HTRE No. 2 with inserts 1-C and 2-A mockups.	Evans (1958a), Flagella (1962)
10/2/2B	12/12/57-3/6/58	12/20/57	NAV	155	NAV[b]	Testing of uncoated BeO ceramic fuel tubes and evaluation of radioactive material releases.	Flagella (1962), Foster et al. (1958) Evans (1958a), Evans (1960b)
11/2/1C	3/12/58-4/14/58	3/20/58	1312	100	141[b]	Testing of unclad, slotted, hydrided zirconium moderator.	Evans (1958b), Flagella (1962)

Table A-24. (continued)

IET Test # HTRE No./ Insert Number	IET Test Dates	First Date of Nuclear Operation	Reactor Energy Production (MW-h)	Hours at Test Conditions	Total Test Hours	Brief Test Description	References
12/2/1D	4/21/58-5/7/58	4/30/58	26.7	0.83	4.9[b]	BOOT Test. Conducted to show effects of reduced air flow and fuel melt.	Devens et al. (1958), Baker et al. (1959), Flagella (1962)
13/3/NA	11/4/58-11/18/58	11/4/58	0.3[d]	NAV	NAV	Critical experiments for the HTRE No. 3 reactor assembly and nuclear excursion.	Devens (no date), Wilks (1959), GE (1959)
14/2/L2A1	3/27/59-5/20/59	4/24/59	1357	100	132[b]	Testing of uncoated BeO ceramic fuel tubes and evaluation of radioactive material releases.	Boone et al. (1959), Flagella (1962), Pincock (1959), Evans (1960b)
15/2/L2C1	5/27/59-6/24/59	6/3/59	976	81	112[b]	Endurance testing of metallic, Cr-UO_2-Ti concentric-ring fueled insert	Evans (1959), Flagella (1962), Boone et al. (1959)
16/3/NA	9/2/59-10/9/59	9/15/59	95	NAV	NAV	Initial power operation of the HTRE No. 3 assembly conducted to define nuclear and physical characteristics.	Miller et al. (1960), Showalter (1959)
17/2/L2E1	10/12/59-12/12/59	11/2/59	1492	106	164[b]	Testing of alumina-coated BeO ceramic fuel tubes and evaluation of radioactive material releases.	Evans (1960a), Evans (1960b), Pincock (1960e), Flagella (1962)
18/3/NA	12/23/59-2/8/60	12/31/59	4855[d]	126	167	Second power operation of the HTRE No. 3 assembly to define nuclear and physical characteristics.	Highberg et al. (1960)
19/2/L2E3	2/9/60-4/30/60	2/17/60	3025	204	271[b]	Testing of zirconia-coated BeO ceramic fuel tubes and evaluation of radioactive material releases.	Pincock (1960d), Flagella (1962), Evans (1960b)

Table A-24. (continued)

IET Test #/ HTRE No./ Insert Number	IET Test Dates	First Date of Nuclear Operation	Reactor Energy Production (MW-h)	Hours at Test Conditions	Total Test Hours	Brief Test Description	References
20/2/L2E2	5/1/60-6/13/60	5/14/60	1831	145	170[b]	Testing of alumina-coated BeO ceramic fuel tubes and evaluation of radioactive material releases.	Flagella (1962), Foster et al. (1960), Evans (1960b), Myers (1960)
21/2/L2A2	6/20/60-8/8/60	6/29/60	348	48	87[b]	FEET #1. Evaluation of the effectiveness of the electrostatic precipitator.	Flagella (1962), Pincock (1960b)
22/2/L2E4	8/12/60-8/25/60	8/22/60	9.57	0.18	3.80	LIME. Evaluation of ceramic fuel melting and consequent fission product releases.	Pincock (1960c), Flagella (1962)
23/2/L2A2	9/1/60-10/14/60	9/7/60	258	NAV	85	FEET #2. Further evaluation of the electrostatic precipitator	Pincock (1960b), Flagella (1962)
24/2/L2E5	10/7/60-10/26/60	10/21/60	25.4	2.08	3.15[b]	LIME-II/Sub-Lime. Evaluation of limited air flow on ceramic fuel and radioactive releases.	Pincock (1960a), Baker (1961), Flagella (1962)
25/3/NA	11/15/60-12/16/60	11/16/60	3571[d]	NAV	~173	This was the final operation of the HTRE No. 3 assembly. Further evaluation of reactor characteristics.	Highberg et al. (1961)
26/2/L2E6	12/22/60-3/31/61	12/27/60	3250	352	399[b]	Testing of zirconia-coated BeO ceramic fuel tubes and evaluation of radioactive material releases.	Field (1961), Flagella (1962)

The ANP Program was canceled on March 28, 1961

a. Operating time above 200 kW.

b. This value is 1% of total power for HTRE No. 2.

c. NAV = not available.

d. Value of MW-h not in available literature. Value taken from RSAC-4 modeling.

rupture detector had to be functional during reactor power operations, and dose projections had to be verifiable with field measurements when release rates exceeded established limits (Levine, 1959, pp. 3-4). The effluent monitoring data were also used to estimate doses to individuals from test operations and to evaluate the performance of the reactor and test inserts under varying experimental conditions.

Effluent monitoring techniques used during the ANP tests reflected the state of the art and continued to be evaluated and improved throughout the testing program. During the HTRE No. 1 testing, the primary technique for characterizing airborne releases was collecting particulate samples from the effluent stream. During IET #4, special studies were performed to correlate exhaust gas activity with fuel flow rate, reactor power, and fuel plate temperature (Holtslag, 1956) and to characterize particulate and gaseous emissions (Ebersole, 1956). Results of other studies performed during IETs #4, #8, #12, #14, and #15 are referenced and discussed in Boone et al. (1959), including evaluations of improved sampling techniques and equipment.

Data obtained from the spot sampler, charcoal trap samplers, real-time stack monitor, and 76-in. duct Jordan monitor were the principal sources of information for evaluating radioactive emissions during IETs and estimating airborne releases. Other information, including (a) environmental monitoring data and (b) descriptions, photos, and measurements made during inspections of the reactor and inserts after tests, also provided important information concerning releases. The information was used to corroborate the effluent monitoring data. Descriptions of the spot sampler, charcoal trap samplers, real-time stack monitor, and 76-in. duct Jordan monitor are provided in the following sections.

Spot Sampler. The spot sampler was used to periodically collect samples from the airborne effluent to assess total fission product airborne releases during IETs. A brief description of the spot sampling techniques used during IET #20 is provided in Foster et al. (1960, p. 65):

> The total release of fission products is monitored during IET testing by taking short, (approximately 1 minute) samples of the

effluent from a probe at the 80 foot level of the stack. This routine monitoring is done by Health Physics personnel. They collect the effluent sample on a paper filter identical to that used in the continuous stack monitor, and after applying corrections for collection efficiency and sample line loss, they extrapolate the activity counted at age 10 minutes to total effluent activity and use the obtained values to maintain a calibration of the stack monitor.

The spot sampling technique remained essentially the same throughout the testing program. Samples collected during IETs #3, #4, and #6 were reported in terms of measured particulate releases (Thornton et al., 1962). During later tests, methods were developed so that the total fission product release could be related to the spot sample results. Airborne releases from IET #10 were expressed in terms of activity released at age 17 seconds. The releases from IET #11 were expressed as both 17-second-decayed curies and 10-minute-decayed curies; releases for subsequent tests were almost exclusively expressed in terms of 10-minute-decayed curies. Because the sample could not be counted instantaneously after collection, some period of decay was inevitable. The actual decay period was also variable for different samples. To provide a consistent means of comparing the estimated airborne releases from the reactor, sampling results were expressed in terms of curies measured after a common period of decay. Samples were corrected for a common decay period by using the assumption that the decay of gross fission products followed a single power-function relationship.

More detail concerning spot sampling methods and equipment are provided in Boone et al. (1959). This reference contains an extensive discussion and evaluation of the spot sampler, with emphasis on the sampling performed in support of IETs #14 and #15. A schematic drawing indicates that the sample was drawn through a 0.5-in. diameter, 110 ft long stainless steel line; through the sample filter; through a flow meter (0 to 6 scfm); through a constant flow regulator; and into a vacuum pump. Gauges were provided for determining the temperature at the flow meter and the vacuum of the sample line at the flow meter and on both sides of the constant flow regulator. A solenoid valve was actuated by a timer to help ensure accurate control of the sampling time (Boone et al., 1959, p. 32).

Several parameters affecting the performance of the spot sampler were evaluated (Boone et al., 1959), including the infiltration of air through the porous fire-brick liner of the stack, sample line losses (plate-out), filtering efficiency, and counting efficiency. A sample line-loss factor of 1.56 had been determined during IETs #4 and #8 "for old fission products artificially injected or boiled out of moderately damaged metal fuel elements" (Boone et al., 1959, p. 7). During IETs #14 and #15, a line loss factor of 1.04 was measured. Of particular importance was the discussion of filtering efficiency and the observation that the filtering efficiency was dependent upon the jet engine fuel flow and resulting combustion products in the effluent stream (Thornton et al., 1962, pp. 26-27 and 181-182). Boone et al. (1959, pp. 8-10) discusses a technique to compensate for the variable filtering efficiency using the ratio of the activity measured on the real-time stack monitor filter paper and the gamma-ray activity observed in the 3-L gas chamber down stream of the stack-monitor filter paper.

The estimates of total fission product releases, made on the basis of the spot samples, often represented the best available information concerning total airborne releases of radioactivity during IETs when evaluated against other sampling results.

Charcoal Trap Samples. Charcoal trap samples were collected from several locations during the IETs, and similar methods were used throughout the testing program. Based on available ANP literature, extensive charcoal trap sampling was first reported for IET #10 (Foster et al., 1958) and continued through the remainder of the ANP Program. Charcoal trap sampling performed during IET #20 is described in Foster et al. (1960, p. 54).

> Effluent samples were collected at several different points during the test. The probe which is referred to as the insert probe is located on the instrument harness directly below the insert in the CTF cocoon. A second insert probe was built into the insert to sample the discharge air from the insert. The sample line from the harness probe is led upward through the shield water through the top plug and thence down the outside of the CTF to the chiller box where the charcoal traps are located. The insert probe sample line was routed thru the upper plenum and thru the top shield and then to a trap located in the CTF chiller box. The probe usually referred to as the vault or stack probe is located at the 80 foot

> level of the stack and samples the effluent at that location. The sample is led down through approximately 100 feet of sample line into the vault located at the base of the stack. The charcoal trap for this probe is located in a bath of running water [and is chilled to approximately 55°F] ... As there are normally personnel in this vault in connection with Health Physics monitoring activities, it is possible to collect more than one effluent sample per run at this location. At the other sampling locations, the charcoal traps can be connected and removed only when the reactor is down; therefore, only one sample per run is obtainable. During the test, an additional probe was installed which samples the effluent in the 76" duct. This trap was chilled by being surrounded with a jacket which contained an ethylene glycol solution which was cooled by mechanical refrigeration.

The sampling techniques used during IET #20 were also described.

> The technique used is the same as has been used in many tests in the past. The charcoal traps are nominally one inch in diameter traps containing a five inch bed depth of 20 - 30 mesh activated coconut charcoal. The effluent samples collected by this technique are taken to the Radio Chemical Laboratory at ITS where they are analyzed for the amount of several specific isotopes contained in the trap (Foster et al., 1960, p. 54).

A schematic drawing of the charcoal trap sampler for the stack is provided in Boone et al. (1959, p. 34). The sample line was 110 ft long and was constructed of stainless steel. The effluent sample was drawn through the sample line, into the charcoal trap, through a flow meter (0 to 1.2 scfm), through a constant flow regulator, and into a vacuum pump. Gauges were included at the flow meter for determining the temperature and vacuum of the sample line.

A sample line loss or plate-out factor of 1.56 was applied to all charcoal trap samples collected from the 80-ft level of the stack before IET #17 (Pincock, 1960e, p. 99). This factor had been determined during IETs #4 and #8 "for old fission products artificially injected or boiled out of moderately damaged metal fuel elements" (Boone et al., 1959, p. 7). A line loss factor of 1.04 was measured for fission products released during IETs #14 and #15 (Boone et al., 1959, p. 7). The fission product release during IET #15 was heavily biased toward the iodines (Boone et al., 1959, p. 26), and

the results of the line loss measurements were directly applicable to evaluating the charcoal trap samples for iodine radioactivity. For continuity and consistency in the analysis of charcoal trap sample data, the larger line loss factor of 1.56 was assumed to apply to samples collected from the 80-ft level of the stack during all IETs.

The results of radiochemical analyses performed on the charcoal trap samples were generally expressed in terms of a leakage rate or release fraction that was defined as ". . . the ratio of the amount of a given isotope released via the effluent during the sampling period to the amount of the same isotope produced in the insert fuel in the same period of time" (Pincock, 1959, p. 92). Mathematically, the leakage rate can be expressed as

$$\alpha = \frac{R}{Q} \tag{A-7}$$

where

α = leakage rate (unitless)

R = radioactivity released during the sampling period (Ci)

Q = radioactivity present at the end of the sampling period as a result of the fissions occurring in the insert during the sampling period (Ci).

Several equivalent terms evolved during the IETs that were used to refer to the leakage rate, including release fraction and fractional release.

Using the leakage rates reported in the GE-ANP literature, release rates for I-131, I-133, and I-135 were calculated in terms of curies per unit time for each run of a test series. These calculations were based on the equations and parameters provided in Foster et al. (1958, pp. 21-24) and Pincock (1959, pp. 92-95) for I-133 and I-135. The release rates for I-133 and I-135 during the sampling period were calculated from the following equation:

$$\frac{R}{t_r} = \frac{\alpha Q}{t_r} = \frac{\alpha \gamma f(1 - e^{-\lambda t_r})}{t_r(2.22E{+}12 dpm/Ci)} \qquad \text{(A-8)}$$

where

t_r = length of the sampling period (minutes)

γ = fission yield of the isotope (either I-133 or I-135) (unitless)

f = fission rate in the test insert (reciprocal minutes)

λ = radioactive decay constant for the isotope (reciprocal minutes).

Because of the complexity of the calculations for I-131, the amount of radioactivity present at the end of the sampling period as a result of the fissions occurring in the insert during the sampling period was estimated using the RSAC-4 computer code. The release rate for I-131 was calculated from the following equation:

$$\frac{R}{t_r} = \frac{\alpha Q}{t_r} = \frac{\alpha q P}{t_r} \qquad \text{(A-9)}$$

where

α = reported I-131 leakage rate based on charcoal trap samples collected during the IETs

q = I-131 activity present at the end of the sampling period as calculated by the RSAC-4 computer code as a result of the fissions occurring in the insert during the sampling period for operation of the insert at a unit power level (Ci/kW)

P = power level of the insert (kW).

A comparison of the calculated release rates for I-131 using the original equations and methods employed during the GE-ANP Program and release rates calculated using the RSAC-4 computer code showed agreement within 2 to 12% over a broad range of sample collection times. Results for nominal sample collection times of 30 to 60 minutes demonstrated agreement within 6 to 10%. This agreement shows that significant biases or inaccuracies were not introduced by estimating I-131 release rates using RSAC-4 calculations.

When sufficient sampling data were available for estimating the I-131, I-133, and I-135 release rates during individual reactor runs of a test series, inventory-normalized release rates were calculated by dividing the isotopic release rate from the last charcoal trap sample collected during a run by the inventory of the isotope estimated to be present in the insert at the end of the run. The insert inventories were calculated using the RSAC-4 computer code. Average values of the inventory-normalized release rates were then calculated for I-131, I-133, and I-135. The largest of these averages, when multiplied by the length of time the reactor operated during a run, provided the halogen release fraction used in the RSAC-4 computer code for estimating the airborne fission product release. In all cases, the largest of the average inventory-normalized release rates were associated with either I-133 or I-135. The halogen release fraction used in the RSAC-4 calculations, therefore, provided a conservative estimate of the amount of I-131 released during a test.

Real-Time Stack Monitor. The real-time stack monitor was used to provide a continuous read-out of the particulate and gaseous fission product releases in the IET airborne effluent. A schematic drawing of the stack monitor is provided in Boone et al. (1959, p. 33). The sample probe was located at the 80-ft level of the stack. The sample was drawn through a 0.5-in. diameter, 110 ft long stainless steel line; through a moving filter paper assembly; through a 3-L gas chamber; through a flow meter (0-2 scfm); through a constant flow regulator; and into a vacuum pump. Vacuum and temperature gauges were provided at the flow meter. The same filter paper was used for the stack monitor and for spot sampling. Two scintillation detectors

were installed under the moving filter paper to provide both an instantaneous readout of the activity collected on the filter and the activity remaining after a period for decay of the short-lived radioactivity. Another scintillation detector was mounted on the gas chamber to measure the amount of gaseous activity released. The inside of the gas chamber was coated with teflon, and water and compressed air were provided for decontaminating the chamber. (Previous uncoated chambers used before IET #14 presented decontamination problems that limited their usefulness.) Preliminary conclusions indicated that the primary application of the gas chamber was to provide a means for continuously evaluating the collection efficiency of the particulate filter (Boone et al., 1959 p. 21).

Seventy-Six-Inch Duct Jordan Monitor. A Jordan-style ionization chamber was located under the 76-in. effluent duct between the reactor and the stack (Foster et al., 1958, p. 28). The instrument was located in a lead shield to maximize the response of the instrument to radioactivity in the duct and minimize the response to radiation from other sources. A continuous read-out of the exposure rate measured by the instrument was available. A rough calibration factor of 10,000 curies per hour per Roentgen per hour was determined during IET #10 based on the correlation of the 76-in. duct Jordan monitor readings with other effluent monitoring data (Foster et al., 1958, p. 28). The usefulness of the duct Jordan monitor was limited by plate-out and deposition of radioactivity within the duct.

HTRE No. 1

A general description of the HTRE No. 1 test assembly was provided in Thornton et al. (1962, p. 21):

> The Heat Transfer Reactor Experiment (HTRE) test assembly as conceived in the basic design consisted of an air-cooled reactor operating a single, modified J47 turbojet engine. The reactor used metallic fuel elements and water moderator. The turbojet engine and shield were part of a mobile facility called the Core Test Facility (CTF). A reactor core, shield plug, control actuators, source rod, startup fission chambers, and operating ion chambers were combined into an integral unit before insertion in the shield.

The reactor structure was aluminum and consisted of a cylindrical water pressure vessel penetrated by air tubes, into which fuel cartridges were inserted for nuclear operation of the reactor. The air tubes were lined with a thin layer of stainless-steel-jacketed, mineral-wool, felt-type insulation to reduce escape of the fuel element heat into the water moderator. The dished-head transition section was an integral part of the reactor assembly in that it was connected to the reactor by control rod guide tubes and water tubes. The reactor and transition assembly were bolted to the bottom of the shield plug through a flange on the transition assembly. The transition section was made of aluminum and contained moderator water for neutron shielding.

The active core was a hexagonal bank of 37 aluminum tubes containing nickel-chromium fuel cartridges with sandwiched UO_2 fuel meat. A length of unfueled water-tube matrix extended on each end of the active core to serve as end reflector. The radial reflector consisted of beryllium slabs arranged in a hexagonal shell.

Fuel for the reactor was supplied by enriched UO_2 mixed with an 80 Ni - 20 Cr alloy in a weight ratio of 40 to 42 percent UO_2. The fuel mixture was clad with a modified nickel-chromium alloy and was fabricated in ribbon form. The fueled ribbon was formed into rings sealed at each end with braze-coated wire. Each fuel element consisted of a concentric arrangement of the fueled rings, joined and spaced at the leading edge by brazed channels, and spaced at the trailing edge by trapezoidal spacers. Eighteen elements, together with the forward ring assembly and the aft assembly, formed the fuel cartridge. The cartridge was divided into two sections on the basis of hydraulic diameter; the first eleven stages formed the first section, and the last seven formed the second section.

The IETs involving HTRE No. 1 are described below:

Three series of tests were performed during HTRE-No. 1 operation. The first series covered the period from December 27, 1955, to February 25, 1956, and was designated Initial Engine Test (IET) No. 3. (Previous tests, designated IET No. 1 and IET No. 2 were not power operations.) The core used in this first test series was called the A2 core and was part of the first test assembly, the D101A2. In the second test series, IET No. 4, which was conducted during the period from April 17, 1956, to June 29, 1956, a slightly modified A2 core was used. After further modification based on test results and a series of shielding tests (IET No. 5), the third series of tests, designated IET No. 6, was performed during the period from September 24, 1956, to January 3, 1957. IET No. 6 employed a completely new reactor test assembly, the D101A3 (Thornton et al., 1962, p. 23).

As indicated, three reactor power operations (IETs #3, #4, and #6) were conducted in the HTRE No. 1 assembly. The reactor operated at power levels up to 20.2 MW and generated about 5500 MW-h of nuclear energy. In general, the airborne releases of radioactivity from IETs #3, #4, and #6 resulted from fuel degradation events such as fuel cladding failure or fuel melting. IETs #1, #2, and #5 did not involve power operations of the reactor.

HTRE No. 2

A general description and summary of the HTRE No. 2 reactor is provided in Flagella (1962, p. 13):

> The development of advanced reactor types for aircraft nuclear propulsion required an extensive in-pile test program of various improved fuel elements and moderators. Many of these tests could not be accommodated in the MTR or other conventional reactor installations because of test specimen geometry, or because of the nature of the test operation. For this reason, the basic HTRE No. 1 reactor design described in APEX-904 was modified to accommodate special test specimens of more advanced fuel element and moderator assemblies. The modified design was designated the HTRE No. 2 Reactor.
>
> The HTRE No. 2 "parent core" was similar to the HTRE No. 1 core, except that the central seven air tubes were removed and replaced by a hexagonal void 11 inches across flats. A corresponding opening was made in the top shield plug so that sections of advanced reactors could be inserted into the HTRE No. 2 parent core without requiring removal of the core from the shield. The inserts were suspended from a small diameter shield plug, which filled the opening in the main shield plug. No special cooling air circuit was provided for the insert. The air was drawn from the common plenum chamber above the reactor.
>
> Since it was expected that some of the inserts would contribute less to reactivity than the seven fuel cartridges which had been removed, an additional four inches of beryllium side reflector was added to the parent core to maintain an acceptable excess reactivity balance.
>
> HTRE No. 2 was used principally for the testing of BeO ceramic fuel cartridges of the type planned for the XNJ140E-1 [nuclear turbojet engine], although some tests were performed using the metallic cartridges and hydrided zirconium moderator of the type used in HTRE No. 3.

As in the design of the HTRE No. 1, cooling air for the HTRE No. 2 was generally supplied by one or two turbojet engines operating fully or partially on chemical power (jet fuel) (Flagella, 1962, pp. 15-18).

Power operations with the HTRE No. 2 assembly began in July 1957 and continued through March 1961. The parent core and inserts were operated at nominal power levels of 8 to 15 MW and generated 16,628 MW-h of nuclear energy during 1353 hours at test conditions (Flagella, 1962, pp. 13-15).

IETs #7 and #9 were critical experiments performed with insert mockups and did not involve power operations of the HTRE No. 2 reactor. Three different metallic inserts (1-B, 1-C, and 1-D) were tested in the HTRE No. 2 reactor in the experiments designated as IETs #8, #11, and #12, respectively. The design of these inserts incorporated hexagonal tubes of clad or unclad zirconium hydride with concentric-ring metallic fuel elements in the holes of the zirconium hydride moderators.

The insert 2-B was used in IET #10 to test ceramic fuels composed of beryllium oxide (BeO), uranium dioxide (UO_2), and yttrium oxide (Y_2O_3). The operation of insert 2-B indicated a need to increase the power density within the test insert. A parent insert, designated as insert L2, was designed with a higher power density and was subsequently used in 10 tests. Nine of these tests (IETs #14, #17, #19, #20, #21, #22, #23, #24, and #26) were performed to evaluate various ceramic fuel designs, and one test (IET #15) involved the evaluation of a metallic concentric-ring fuel element cartridge (Flagella, 1962).

HTRE No. 3

The HTRE No. 3 was built in the configuration of an assembly anticipated to be used in an operational aircraft.

> The design of HTRE-No. 3 ... differs from the first two power plants [HTRE No. 1 and No. 2] in three basic ways: (1) it has a solid moderator, (2) the reactor is horizontal, and (3) a single reactor supplies the heat for two turbojet engines operating simultaneously. Using lightweight hydrided zirconium as a

> moderator instead of water in the HTRE No. 3 eliminated the necessity for circulating and removing heat from the moderator water, reducing weight and complexity, increasing the power-to-weight ratio, and freeing the power plant from the cooling ram-air temperature limitation (Linn et al., 1962, p. 15).
>
> The dimensions of the core and its structural characteristics as well as the design temperatures were those of a power plant capable of providing useful flight propulsion. The power generated by HTRE No. 3 ranged up to 35 megawatts... In the HTRE No. 3 tests, the power levels were so chosen that the fuel element temperatures, the key parameter, would be characteristic of flight service (Linn et al., 1962, p. 15).

Reactor testing with the HTRE No. 3 started in April 1958 and was completed on December 19, 1960. Initial low-power tests with the HTRE No. 3 assembly were conducted at the Low Power Test (LPT) Facility during April, May, and June 1958. The assembly was then moved to the CTF on September 8, 1958, for testing the reactor operating systems and further low-power testing. After a period of initial low-power operations during IET #13, a nuclear excursion occurred on November 18, 1958. The excursion was a result of the dynamic and shim rods being withdrawn by the reactor control system under the influence of an erroneous reactor power indication. Damage caused by the excursion was limited to the fuel in the reactor core (Linn et al., 1962, pp. 39-40).

The reactor core was rebuilt, and on June 4, 1959, the assembly was returned to LPT for low-power testing and comparison with the original core characteristics. The power plant was delivered to the CTF on July 28, 1959, for final checkout before operations testing. Criticality was achieved on September 8, 1959, after which testing continued in IETs #16, #18, and #25 (Linn et al., 1962, pp. 40-42).

<u>IET #1 (dates not available)</u>

Thornton et al. (1962, p. 23) states that IET #1 was not a power operation. Thornton et al. (1962, p. 172) lists a report entitled *Engine and Duct Valve Operational Report IET No. 1*, indicating the testing was performed on nonnuclear operational aspects of the engine and duct valve. Because there was no power operation of the reactor, no radioactive material was released.

IET #2 (dates not available)

Thornton et al. (1962, p.23) states that IET #2 was not a power operation. Thornton et al. (1962) is the only reference available that mentions IET #2. Communication with former GE-ANP employees still employed at the INEL indicated that this test series may have been comprised of critical experiments for the HTRE #1, which logically fits into the probable operating schedule.

IET #3 (February 11 through 24, 1956)

Following the critical testing of the HTRE No. 1 assembly, an operational period of power testing, designated as IET #3, was scheduled to determine the nuclear characteristics of the core, the characteristics of the cooling system and system components, and the control characteristics of the control system. IET #3 was performed between December 27 and February 24, 1956. Releases of airborne radioactivity were modeled as if they had all occurred between February 11 and 24, 1956. As summarized in Thornton et al. (1962, p. 99):

> The first series of operational tests using the D101A test assembly was run at the Idaho Test Station during the period from December 27, 1955 to February 25, 1956. It was designated Initial Engine Test (IET) No. 3. The test operation, 44 runs in all, was generally successful in that the system operated without chemical assistance as intended; no inherent instabilities were observed. On February 11, 1956, during an attempted transfer to full nuclear power, a burst of stack activity was detected by the monitoring equipment. The presence of fission fragments was established during subsequent operation by the presence of I^{131} in the stack gas, an indication that damage to the fuel elements had occurred. The test series was terminated to assess the damage.

The report then discussed the test in detail (Thornton et al., 1962, p. 99):

> Following a series of cold-flow tests on the engines, the reactor was operated first without forced-air cooling to determine heat dissipation of the core and next with various combinations of auxiliary afterheat blowers. . .

> The reactor was first operated at substantial power (above 200 kilowatts) on January 17, 1956, and was operated at powers above this level on 18 days for 40.21 total hours, 349 total megawatt-hours, 16.9 megawatts maximum, and 8.7 megawatts average. . . IET No.3 operations showed that the test assembly was very stable.

During the operational period, six transfers to full nuclear power were attempted; only three transfers were successful. As a result of the successful transfers to full nuclear power, a total of 6.3 hours of this operational period was spent at full nuclear power (Thornton et al., 1962, p. 100).

Fission Product Inventory. The inventory of radioactive fission products in the HTRE No. 1 reactor for IET #3 was based upon a simplified treatment of the operating history. The operating history was recounted in Thornton et al. (1962, p. 100). Inventory calculations were performed with the RSAC-4 computer code. The operation before February 8 was simplified by assuming it was comprised of three periods: 22 hours at 200 kW, 11.03 hours at 8.7 MW, and 17.6 hours at 200 kW, for a total power of 103.9 MW-h. This value is conservative with respect to the actual power generated during this time, which was 95.81 MW-h.

For the 16-day period following February 8, the reactor was conservatively assumed to operate continuously except for the shutdown periods of 24 hours or longer. During this time, the reported energy produced was assumed to be generated at the higher power levels, and for shutdown periods of less than 24 hours, the reactor was assumed to be operating at 200 kW. For the total modeled operation, the calculated energy released by the reactor is about 30 MW-h greater than actually occurred. This conservatism will increase the calculated inventory of long-lived fission products in the core slightly.

Airborne Radioactivity Releases. Radioactive material releases for IET #3 consisted of fission products produced during the operation; uranium, released at the time of fuel degradation; and argon-41, created by neutron activation of stable argon in the cooling air flowing through the reactor.

Fission Product Releases--Airborne fission product releases for this test series are summarized in Thornton et al. (1962, pp. 135-136):

> The release of radioactive material during IET No. 3 was first detected February 11, 1956, during an attempted transfer to full nuclear power. Fuel cartridge damage was suspected and later verified, during disassembly of the A2 core, as the cause.
>
> The presence of fission products in the exhaust was definitely proved later in the test series when I^{131} was found in the particles carried out of the stack during the second 100 percent nuclear operation. The radioactive material released from the stack during this operation was estimated at 2000 curies over a 4-hour period. An AEC site-survey crew could find no trace of this radioactivity, even though they surveyed the area during and after the operation. This lack of evidence of radioactivity was probably caused by a head-on meeting of opposing winds, which caused an upward flow and dispersed the radioactivity to the mountains.
>
> The measured radioactivity released from the stack during the third 100 percent nuclear power operation was about 1000 curies over a 2-hour period. There was some fallout from this operation at the IET area, the ANPD Administration area, and the A and M area, but not enough to seriously limit use of any of these areas. After the reactor was shut down, the AEC site-survey crew found some fallout at the ANPD main gate and along an AEC highway for 3 miles south of the ANPD main gate. None was found on any public highway.
>
> Much smaller amounts of fission product radioactivity were released from the stack on succeeding days.
>
> During the last day of operation, an attempt was made to verify the location of the damaged fuel cartridge by observing the released radioactivity when control rods proximate to suspected tubes were withdrawn. The comparative activity levels of the rupture-detecting filters had indicated that tubes 26 and 30 were the most radioactive. A short run at about 60 percent nuclear power was made to locate the damaged fuel element. The wind was from the southwest at 30 miles per hour, and the Idaho site-survey crew was located downwind from the IET. The survey crew radioed that they were picking up a maximum air activity of about 1 mr per hour on the Salmon Highway and at Monteview but that most readings were near zero. At the request of the Idaho Operations Director of Health and Safety the operation was continued to allow the survey crew to get a better air sample. A short while later the control rods adjacent to tube 30 were pulled, and both the rupture detector and the stack monitor indicated a slight burst of activity. Twenty minutes later a portion of the monitoring crew located in Monteview, 10 miles away, detected some activity, apparently a result of this burst. At this point, with the

concurrence of the Idaho Operations Director of Health and Safety, the reactor operation was terminated. About 100 curies was released during these tests. The power plant was still operating satisfactorily at this time.

Based on this discussion of releases, the total amount of radioactive material released to the environment was conservatively estimated to have been approximately 4000 10-minute-decayed curies of fission products.

Release of fission products occurred primarily from February 11 through 24, with over 90% of the total activity released on February 13, 18, 22, and 24. Fission product releases on February 11 and 24 were added to those on February 13, when 48% of the total fission product releases occurred.

The release fractions for IET #3 were estimated based on the fuel damaged shown in photographs taken when the reactor was dismantled following the test (Thorton et al., 1962, pp. 137-141). The release fractions chosen for the RSAC-4 computer code resulted in a total estimated release of 42,550 10-minute-decayed curies. This activity is equivalent to the fission products contained in 1.63 fuel assemblies with fractionated releases of 1.0 for the noble gases, 0.5 for the halogens, and 0.01 for the solids. In terms of the total core inventory the release fractions were 0.044 for the noble gases, 0.022 for the halogens, and 0.0044 for the solids.

Uranium Releases--Postoperational examination of the core following the test series showed that considerable damage had occurred to the fuel because of cladding failure. Not all of the uranium from the damaged fuel would have been released to the environment because of recongealing of the molten material at regions outside the core and plateout and deposition of material in the 76-in. duct that leads to the IET stack. As discussed above, 0.0044 of the total core solids were released. The same release fraction was applied to the uranium in the fuel. The core contained 37 fuel assemblies, each containing 1100 grams of fuel (Thornton et al. 1962, p. 46). Therefore, 0.0044 of the 40700 g of fuel in the core, or 179 g of uranium, was released. By the methodology described by Rich et al. (1988, pp. 2-7 to 2-10), the U-234, U-235, and U-238 activity in this mass of uranium was determined. The

composition of the 179 g of 93.4% enriched uranium assumed to be released was 1.14E-2 Ci of U-234, 3.62E-4 Ci of U-235, and 3.36E-6 Ci of U-238.

Argon-41 Releases--In addition to the release of fission products and uranium during the reactor operation, Ar-41 was also released. This radionuclide, created by the neutron activation of stable argon in the air that passes through the core, is dependent on the natural abundance of Ar-40 in air and the thermal neutron flux to which it is exposed. The calculation [basically the same as presented by Gamertsfelder (1952, p. 95), except for the different value of 0.65 barns for the thermal neutron absorption cross section of the calculation used here] shows that Ar-41 was produced in the HTRE No. 1 core at the rate of about 2.8 Ci/MW-h. The Ar-41 release was estimated to be 2250 Ci based on the assumption that the reactor operated at a power level at 20 MW for the entire 40.21 hours of testing during IET #3.

IET #4 (May 1 through June 29, 1956)

IET #4 was performed between April 17 and June 29, 1956. Releases of airborne radioactivity were assumed to occur between May 1 and June 29, 1956. The second series of operational tests is described in Thornton et al. (1962, p. 143).

> The second series of operational tests using the D101A test assembly was run at the Idaho Test Station during the period from April 17, 1956, through June 29, 1956, and was designated IET No. 4. The primary purpose of the tests was to determine whether the modifications based on the results of the first test series had significantly improved the capabilities of the reactor. Additional objectives were (1) to make complete measurements of the power-plant performance, (2) to measure xenon poisoning, and (3) to study and improve servo control of the reactor.
>
> IET No. 4 utilized the A2 core in which several significant repairs and modifications were made as a result of IET No. 3 operation. Thirteen new fuel cartridges with extra rails were installed. Fifteen control rods were replaced.
>
> A third major modification entailed redesign of the insulation sleeves to provide more assurance against liner collapse and subsequent fuel cartridge damage. The insulation sleeve was modified as follows:

1. Thinner liners were used, and Thermoflex insulating material was removed from the tail sections.

2. Thirty six air bleed holes were incorporated through the insulation tube to prevent pressure buildup between the core tube and insulation sleeve.

3. A stiffening ring was added in the insulation tube at the rear of the eighteenth fuel stage to provide strength against collapse.

4. The diameter of the cartridge tail assembly was reduced to prevent the occurrence of an air seal at the rear of the insulation tube.

5. The insulation tube slip joint was removed to prevent collapse of the tube along this line.

The core was loaded with 24 cartridges used during IET No. 3 and 13 new cartridges. . . During this test series, the reactor was operated for a total energy release to the air of 1877 megawatt-hours at a maximum sustained power level of 16.0 megawatts to air. The maximum sustained plate temperature recorded was 1991°F, with a maximum sustained average of 1701°F. The maximum core discharge temperature was 1394°F. The total operating time at a power to air of 16 megawatts was 84 hours.

Table A-25 contains the reactor operating history during IET #4 to further explain how the operation was modeled for the RSAC-4 computer code.

As indicated by operations in Table A-25, the operational period can be broken into three periods: (1) relatively low power levels, (2) higher power levels, and (3) near the end of operation where fuel damage was assumed to have occurred. Based on the documented discussion of radioactive material releases and the operational history of the reactor, the releases for this test series were chosen to occur uniformly over the entire period of low-power operation (May 1 to May 23, 1956); the entire period of higher-power operation (May 24 to June 26, 1956); and on the date of the last operation, when fuel damage occurred (June 29, 1956).

Fission Product Inventory. The fission product release for the first two operational periods was modeled based on the generation rate of fission products in the reactor core. The fission product release during the third

Table A-25. IET #4 reactor operation

Date	Time Above 200 kW (h)	Maximum Power of Total System (MW)	Total System Power (MW-h)
5/1/56	2.52	4.1	3.22
5/2/56	6.78	8.1	34.20
5/3/56	1.90	4.4	7.25
5/4/56	4.37	9.6	35.06
5/5/56	4.47	11.2	36.68
5/7/56	4.82	11.2	51.31
5/9/56	2.00	11.2	15.60
5/10/56	4.05	8.4	16.37
5/14/56	4.66	11.4	49.61
5/15/56	0.13	0.8	0.10
5/16/56	3.27	3.2	5.59
5/17/56	3.48	9.9	27.28
5/18/56	1.88	6.4	3.65
5/19/56	2.28	13.4	25.82
5/22/56	7.67	13.6	67.63
5/23/56	8.92	13.8	61.68
5/24/56	6.70	14.6	49.03
5/26/56	9.47	16.1	73.36
5/31/56	9.17	16.3	131.70
6/1/56	10.06	17.1	158.31
6/5/56	10.02	17.6	157.96
6/6/56	6.43	18.1	95.75
6/7/56	7.20	17.2	108.80
6/8/56	1.92	15.5	20.03
6/9/56	3.95	16.5	36.28
6/12/56	7.10	16.7	71.01
6/13/56	8.83	16.6	125.64
6/14/56	10.45	16.3	149.57
6/16/56	0.78	1.8	0.92
6/19/56	0.57	13.3	2.98
6/20/56	7.72	18.4	53.51
6/21/56	7.85	17.8	109.08
6/23/56	4.87	17.1	72.57
6/26/56	6.92	17.7	87.67
6/29/56	8.73	16.8	119.76
TOTAL			
IET #4	193.94		2064.98
IETs #3 and #4	234.15		2414.06

period of operation was modeled as a release of a portion of the fuel in the core and the associated fission products.

As shown in Table A-25, the operation above 200 kW spanned a noncontinuous time period of 60 days. Of this period, the reactor operated above 200 kW for a total of 193.94 hours (when the values in the table are added, the total is 191.94 hours). To model this operation with the RSAC-4 computer code, the operation was divided into three periods. The first period, from May 1 through May 23, 1956, corresponded with the time period when the average reactor power level was 10.2 MW. The second period, from May 24 through June 26, 1956, corresponded to a period of higher power operations when the power level was 16.8 MW. The third period of operation, June 29, 1956, corresponded to the end of the power operation when fuel failure was assumed to occur and fuel that contained fission products was released to the environment. The third period of operation was modeled to produce a fission product inventory that would have been present in the HTRE fuel at the end of the operation. To model the operating history, actual time periods were used for the last 17 days of operation, and the assumption was made that when the reactor was not at operating power level, it was operating at 200 kW. For the period before June 12, 1956, the reactor operation was assumed to be constant at a power level of 1.325 MW for 40 days. Modeling the operational history in this manner provided a fuel burnup that was slightly in excess of the documented burnup experienced during the IET #4 operational sequence (i.e., 2141.8 MW-h for the modeled operational history versus the documented value of 2064.98 MW-h). For all three periods of time, the average power level was conservatively calculated assuming that the reactor operated at a power level equal to the maximum power for the entire time that the reactor was operated at power levels above 200 kW.

Airborne Radioactivity Releases. The radioactive material releases during IET #4 were assumed to be fission products generated during the reactor operation, uranium resulting from fuel degradation, and Ar-41 generated by activation during the reactor operation.

Fission Product Releases--Fission product releases during IET #4 were described in Thornton et al. (1962, p. 26).

> From the start of power operation, radioactivity was observed on the stack monitor and rupture detector. Tests to pinpoint the cause of this release indicated that the measured particulate activity did not depend significantly upon fuel element plate temperature at low temperature levels but increased sharply at high plate temperatures. The activity showed a moderate dependency of fuel flow, since it decreased when the fuel flow was reduced. The effect of power on activity was not detectable at low powers, but showed a minor increase at the maximum power tested, 15.5 megawatts. It was therefore concluded that fuel element temperature level was the most significant parameter in determining release of radioactivity.

Because the particulate activity released was dependent on the fuel element plate temperature, data from the test illustrating this dependence (Thornton et al., 1962, p. 166) was graphed in Figure A-3 to show release versus plate temperature. Figure A-3 was used with the time-weighted power levels to determine the release for a given time period. For calculational purposes, it was assumed that there was a direct correlation between the fuel plate temperature and the power level of the reactor. For example, a plate temperature of 1400°F is assumed to be equivalent to a power level of 14 MW.

Because radioactive material was released from the start of power operation, the assumption was made that the material was being released at some fraction of the generation rate applicable to the power level of the reactor. Table A-26 shows the generation rate of fission products at a reactor power level of 10.2 MW applicable to the whole core.

Table A-26. Generation rate of fission products during IET #4 at a reactor power level of 10.2 MW

Fission Product Group	Production Rate (Ci/s)
Solids	4550
Halogens	134
Noble gases	770
Cesiums	426
Rutheniums	15.2

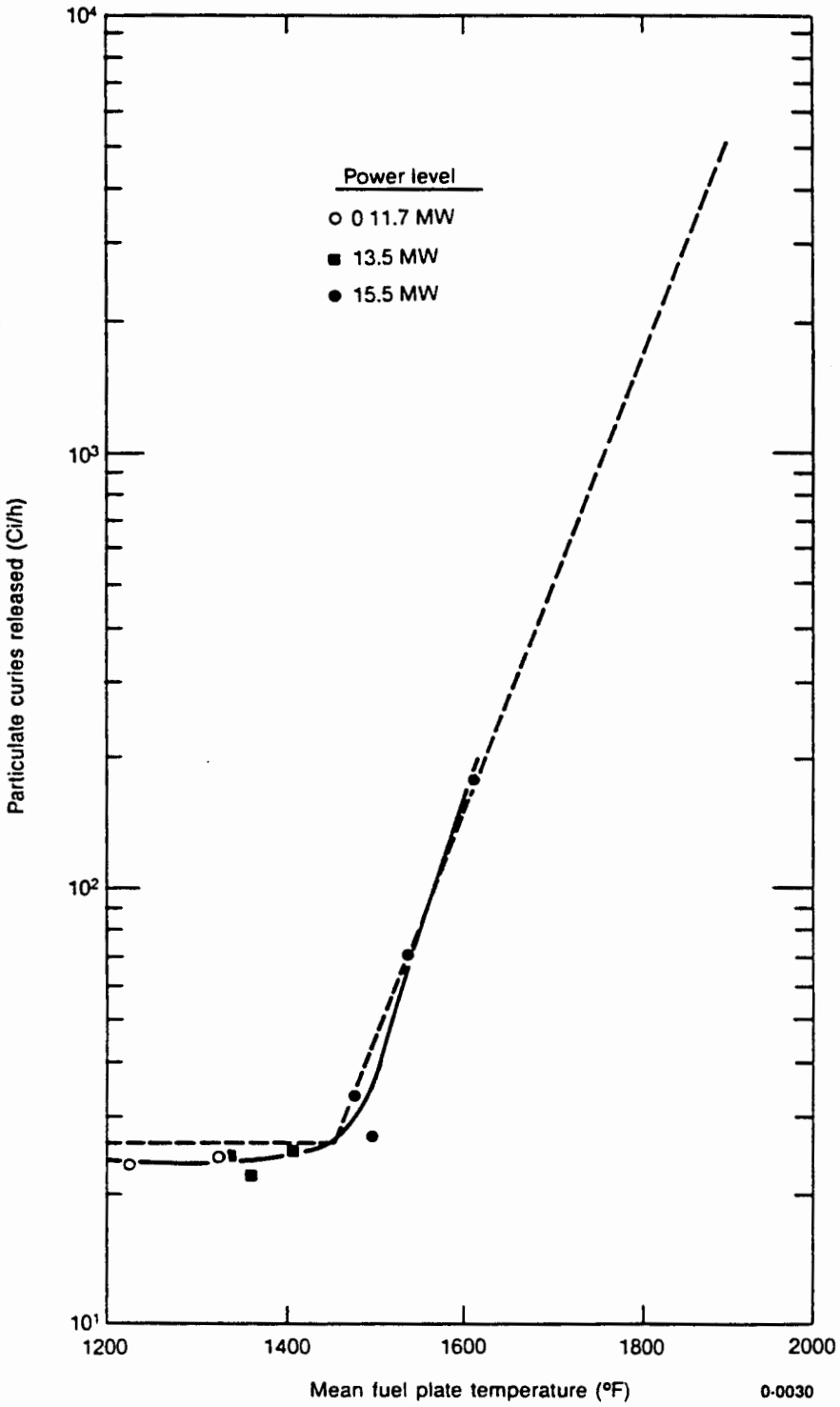


Figure A-3. IET #4 quantity of particulate activity.

The generation rate at 16.2 MW is directly proportional to the power levels (i.e., 16.2/10.2 or 1.59 times the production rate presented in Table A-26).

These production rates were applicable to the core at any time, assuming that the power level of 10.2 MW was maintained. If the release rate was based on the initial generation rate provided above, the released percentage of I-131 is slighted. However, if the release rate is based on an accumulated amount of fission products, as would be the case when it took a finite time for the fission products to migrate through the fuel, then the percentage of I-131 would be greater. Therefore, in an effort to increase the contribution from radionuclides with longer half-lives and larger dose conversion factors, the release was assumed to originate from fission products present after 4 hours of reactor operation.

Several assumptions were made to conservatively estimate and maximize the fission product release for these calculations. The first assumption was that during the first and second of the three time periods analyzed, the reactor operated at 10.2 and 16.8 MW, respectively, for the time above 200 kW. A second maximizing assumption is that although the halogens were undoubtedly a constituent of the solids, for release purposes, the particulate release rate was assumed to contain only the particulate solid fission products and no halogens. The iodines were assumed to be released in addition to the solids. The release was also assumed to originate from the inventory of fission products present after 4 hours of operation during the first and second periods. The release fractions of each of the groups of other solid fission products (cesiums and rutheniums) were chosen to be the same as for the solids.

Because radioactive material was assumed to be released as generated, the material had to be fractionated so that for the first period, when the operating power level was 10.2 MW, the release rate for the solids amounted to 26 Ci/h (10-minute-decayed curies) (see Figure A-3). This release rate was assumed to occur during the total 63.2 hour first period of operation, thus, releasing 1643 Ci. For the second time period, when the power level was assumed to be 16.8 MW, the release of solids was assumed to be 405 Ci/h

(10-minute-decayed curies) (see Figure A-3) for the total 128.74-hour period that the reactor was assumed to operate at 16.8 MW. On the basis of engineering judgment, the RSAC-4 computer code was programmed so that the release fraction for the noble gases during the first and second periods amounted to 2000 times the release fraction of the particulates and two times the release fraction for the iodines. At the end of IET #4 operation, the third period, when fuel degradation was assumed to occur, fuel and associated fission products were released to the environment. Based on pictures of the fuel assemblies taken after the final period of operation, it was estimated that 1.27% of the core fuel and fission products were released to the environment. This quantity is equivalent to the release of 47.1% of one of the 37 fuel assemblies. All release quantities are shown in Table A-27.

Table A-27. Fission product and fuel quantities released to the environment

	Total Release by Group (Ci)				
Decay Time (s)	Solids	Halogens	Noble Gases	Cesium	Ruthenium
Period 1:	**Ar-41: 2.8 Ci/MW-h x 10.2 MW x 63.2 h = 1805 Ci**				
0.0	7.73E+1	8.45E+3	2.72E+4	8.69E+0	2.36E-1
600.	9.37E+2	2.95E+3	6.81E+3	7.09E+2	1.67E-1
3960.	1.14E+3	1.77E+3	2.69E+3	6.18E+2	1.33E-1
Period 2:	**Ar-41: 2.8 Ci/MW-h x 16.8 MW x 128.74 h = 6056 Ci**				
0.0	2.45E+3	2.68E+5	8.62E+5	2.75E+2	7.47E+0
600.	2.97E+4	9.36E+4	2.16E+5	2.25E+4	5.29E+0
2520.	3.68E+4	6.82E+4	1.07E+5	2.71E+4	4.49E+0
Period 3:	**Ar-41: 0.0 Ci**				
0.0	4.80E+4	5.51E+3	8.36E+3	5.03E+3	2.11E+2
600.	2.05E+4	2.48E+3	2.49E+3	1.71E+3	1.70E+2
1080.	1.70E+4	2.39E+3	1.99E+3	1.36E+3	1.61E+2

Uranium Releases--In addition to the fission products released during the third period, it was assumed there was 520 g (47.1% of one fuel tube) of 93.4% enriched uranium fuel released to the environment. Radiological surveys of the engine and the 76-in. diameter ducting to the stack showed that not all of the fuel was released to the environment. However, 520 g of fuel was conservatively assumed to be released.

Argon-41 Releases--During IET #4, 7861 Ci of Ar-41 was released to the environment. This quantity is based on a generation rate of 2.8 Ci/MW-h at the reactor power levels and for the periods of time shown in Table A-27.

IET #5 (dates not available)

Thornton et al. (1962, p. 23) states that IET #5 was a series of shielding tests. The test apparently did not involve power operation of the reactor; however, no specific report about the IET #5 series could be located.

IET #6 (December 18, 1956)

IET #6 was the last nuclear test conducted in the HTRE No. 1 assembly. IET #6 was performed between September 24, 1956, and January 3, 1957. Releases of airborne radioactivity were modeled as if they had all occurred on December 18, 1956. The release of radioactive material during this test series was considerably lower than for previous tests conducted in the HTRE No. 1 reactor. The test consisted of a new core assembly and new fuel elements; it differed from the A2 core because it had a new insulation sleeve design as described in Thornton et al. (1962, p. 171).

> The new sleeve employed a helical winding technique and stiffening rings that enhanced the structural integrity of the sleeve against pressure collapse . . .. This design resulted from an intensive development effort performed to determine the cause of fuel cartridge damage and to prepare for operation of the A3 reactor.
>
> The immediate objectives of IET No. 6 test series were to:
>
> 1. Evaluate the performance of the redesigned insulation liners.

2. Extend and supplement IET No. 3 low-flow, no-flow, and nuclear characteristics.

3. Verify the xenon characteristics determined during IET No. 4.

4. Continue basic controls investigations.

5. Conduct endurance testing with the engine on full nuclear power.

The reactor was first made critical on October 3, 1956, and exceeded 200 kilowatts or 1 percent power on October 12, 1956. During IET No. 6, the reactor was transferred to full nuclear power 40 times and operated for a total energy release to air of 2811 megawatt-hours and a maximum sustained power level of 18.4 megawatts to air (Thornton et al., 1962, p. 173).

Fission Product Inventory. The fission product inventory for this test was generated in a simplified manner similar to that method used for IETs #3 and #4. For this inventory calculation, the documented energy generated for this test series was preserved. The operation assumed for programming the RSAC-4 computer code consisted of a relatively long, 650-hour operation at a power level of 3.63 MW, followed with an immediate operation for 28.66 hours at 18 MW. For IET #6, the majority of the release occurred on December 18, 1956.

Airborne Radioactivity Releases. The fission product releases for this test series were relatively small compared to the other HTRE No. 1 tests. For the purposes of dose calculation, the releases were conservatively accumulated during the operation and were assumed to be released on December 18 over a 8-hour period. This release consisted of the Ar-41 that was generated during the entire operation of 3092 MW-h, the fuel that was conservatively assumed to be released to the environment during the minor fuel degradation that occurred, and the fission products that were calculated to be associated with the fuel that was released.

Fission Product Releases--The release rate of radioactive material during IET #6 was only about 3% of the release rate that occurred during IET #3. The largest release rate for IET #6 occurred during the evening of December 18, 1956, and amounted to 25 Ci/h of particulate activity (Thornton

et al., 1962, p. 183). It was assumed that the minor fuel element damage that occurred during this test operation (Thornton et al., 1962, p. 194) occurred at this time and resulted in the release of a conservatively estimated 5 g of fuel and associated fission products. Five grams of uranium represented 0.0122% of the fuel and fission products in the reactor. This value was applied in the RSAC-4 computer code to all fission products for calculating fission product releases. At the time of this incident the reactor had generated 2875.4 MW-h of energy (Thornton et al., 1962, p. 174).

Uranium Releases--As indicated in the discussion of Fission Product Releases, a release of 5 g of 93.4% enriched uranium was estimated to have occurred based on the description and pictures of the damaged fuel (Thornton et al., 1962, pp. 189-196).

Argon-41 Releases--A total of 8050 Ci of Ar-41 was estimated to have been released during IET #6 on the basis of a production rate of 2.8 Ci/MW-h.

IET #7 (February 22 through July 1957)

IET #7 was a series of critical experiments performed to determine the nuclear characteristics of the HTRE No. 2 reactor and at least two proposed test insert mockups. The nuclear characteristics studied included the power distribution, gamma-flux distribution, and reactivity of the reactor and proposed test inserts. *Engineering Progress Report No. 23* for the time period from January 1 through March 31, 1957, provides background information on the critical experiments and an approximate date for the start of the experiments:

> The initial criticality experiment dolly, carrying the A4 reactor [HTRE No. 2], was received at the IET facility [during the quarter].
>
> Operation of the initial criticality experiment was begun during the quarter, and the initial criticality of the A4 core was accomplished using nuclear mockups of the two insert assemblies. Power mapping and reactivity experiments were begun and were about 40 percent complete at the end of the quarter (GE, 1957, p. 31).

> The HTRE No. 2 with the nuclear mockup of Insert No. 1 was made critical about the middle of the quarter (GE, 1957, p. 35).

The dates on which critical experiments began are provided in Evans (1957b, p. 4):

> The HTRE #2 with insert #1 was made critical on Friday, February 22, 1957.
>
> The HTRE #2 with insert #2 mockup (beryllium in place of BeO) was made critical on Monday, February 25, 1957... An experiment on February 26, in which a fraction (approximately 1/12) of the insert #2 mockup beryllium was replaced with BeO indicated, through extrapolation, that a BeO insert in the HTRE #2 would produce a Kex of 5.0 $\pm$ .5 percent.

The critical experiments were completed before July 18, 1957, when the power operations of IET #8 were initiated (Evans, 1957a, p. 55).

The only explicit reference to IET #7 in the test literature reviewed was by Evans (1958a, p. 30). The discussion of results of critical experiments for insert 2-A includes Figure 1.8-2, "Relative Power Distribution, IET-7 Insert 2-A."

As indicated previously, the critical experiments were performed with nuclear mockups of the test inserts. The mockup of insert No. 1 was designated as insert 1-A, and the insert 2 mockup was designated 2-A. These mockups were not designed for power operations in the reactor; hence, they were operated only very low power levels. A major portion of the critical experiment program with the A4 core and insert 1-A was devoted to determining power distributions, with typical reactor experiment run times of 20 minutes (Evans, 1957a, p. 5). The total power generated during a base run experiment lasting 20 minutes was 6 W (Evans, 1957a, p. 40). This power level was more than a million times smaller than those encountered during the power testing of the insert. No airborne radioactive emissions were assumed to have occurred during the low-power critical experiments of IET #7.

IET #8 (July 31 through August 28, 1957)

IET #8 was performed between July 18 and August 28, 1957, and was the first power operation of the HTRE No. 2 reactor. Releases of airborne radioactivity were assumed to occur between July 31 and August 28, 1957, based on a reconstruction of the operating history of the test. This test series involved the evaluation of the insert 1-B.

> Insert 1B consisted of seven stainless-steel-clad, hydrided zirconium, hexagonal tubes as moderating material with concentric ring metallic fuel elements in the hole of the moderator. This insert, a model of the section from the HTRE No. 3 core, served as the basis from which most of the nuclear design of the HTRE No. 2 was predicted. The primary purpose of the insert 1B tests was to evaluate clad hydrided zirconium as a moderating material capable of withstanding temperatures of 1650°F (Flagella, 1962, p. 39).
>
> The first significant power levels were achieved on July 25th and a maximum total reactor power of 11.4 megawatts on August 2nd. At the conclusion of the power testing on August 27, a total of 33.97 hours had been accumulated at maximum power conditions. During this period, no fission fragment evolution was measured, however, the evolution of molybdenum and manganese and other radioactive particles were detected through the examination of filter papers. Upon disassembly of the reactor, preliminary examination has shown molybdenum and manganese deterioration of the outer cladding of insert tube 3 (Evans, 1957a, p. 55).
>
> Tests were concluded because it was deemed important to determine the source of the apparently high molybdenum and manganese evolution (Evans, 1957a, p. 89).

It should be noted that reactor operations continued through August 28 (Evans, 1957a, pp. 84-85) even though power testing concluded on August 27.

Assumed Reactor Operations. A comprehensive summary of reactor operations was not available for IET #8. Total reactor operating times were estimated from the repeated measurements of excess multiplication (K_{ex}) made throughout the test series (Evans, 1957a, pp. 81-84). The times during which the reactor operated at a power level in excess of 10 MW were taken from graphs of insert temperature and power levels for individual test runs, when available (Evans, 1957a, pp. 61-76). The reactor was assumed to operate above 10 MW for 4 hours during each of the test runs from August 19 through 28, in accordance with the general test practices described by Evans (1957a, p. 56).

No operations above 10 MW were assumed for the apparently short periods of operation on August 8 and 15. The dates and operating times used to evaluate the airborne emissions from IET #8 are shown in Table A-28. The total period of reactor operation at power levels greater than 10 MW was estimated to be 54.5 hours, which exceeded the 33.97 hours of operation at maximum power reported in Evans (1957a, p. 55).

Airborne Radioactivity Releases. The radioactivity released during IET #8 was assumed to be composed of radioactive activation products of molybdenum and manganese, noble gas fission products, and Ar-41. The release of activation products was based on reported and assumed releases during the test. A release of fission products, consisting only of noble gases, was assumed to occur during those periods when the reactor was operated at power levels exceeding 10 MW. No uranium releases were assumed to have occurred during IET #8. Argon-41 was produced in proportion to the energy produced by the reactor during each period of operations.

Activation Product Releases--Only qualitative information is available concerning the composition of the particulate radioactivity measured in the airborne effluent during IET #8. The statement was made that "no fission fragment evolution was measured, however, molybdenum and manganese and other radioactive particles were detected through examination of filter papers" (Evans, 1957a, p. 55). The hydrided zirconium moderator in the insert was clad with molybdenum and stainless steel (GE, 1957, pp. 31, 33, and 111), and the observed particulate activity was derived from a degradation of this cladding (Evans, 1957a, pp. 55 and 89). Neutron activation of molybdenum and manganese in the cladding would have produced Mn-56, Mo-93m, Mo-93, Mo-99, and Mo-101. It is likely that these radionuclides accounted for the majority of the particulate radioactivity released. Consequently, subsequent calculations assume that the particulate releases during IET #8 were composed entirely of the radioactive activation products of molybdenum and manganese.

Particulate radioactivity releases from August 16 to 28 were recorded in Evans (1957a, p. 85) and totaled 153 Ci. For reactor operations conducted from July 31 through August 9, a nominal particulate release rate of 5 Ci/h,

Table A-28. Assumed operating times and radioactive releases for IET #8

Date	Operating Time[a] (h)	Operating Time Above 10 MW (h)	Activation Product Release (Ci)	Noble Gas Release[e] (Ci)	Ar-41 Release[f] (Ci)
7/31/57	7.	4.5[b]	22.5[d]	9.0	149.
8/02/57	7.	5.[b]	25.[d]	10.	165.
8/06/57	7.75	4.[b]	20.[d]	8.0	132.
8/08/57	1.	0.	0.[d]	0.	0.
8/09/57	2.	4.5[b]	22.5[d]	9.0	149.
8/15/57	3.	0.	0.[d]	0.	0.
8/16/57	5.75	4.5[b]	27.	9.0	149.
8/19/57	4.75	4.[c]	28.	8.0	132.
8/20/57	7.5	4.[c]	30.	8.0	132.
8/21/57	6.25	4.[c]	12.5	8.0	132.
8/22/57	4.75	4.[c]	17.	8.0	132.
8/23/57	5.75	4.[c]	18.7	8.0	132.
8/26/57	6.25	4.[c]	2.5	8.0	132.
8/27/57	9.25	4.[c]	10.	8.0	132.
8/28/57	5.	4.[c]	7.5	8.0	132.
TOTAL	83.	54.5	243.	109.	1800.

a. Operating times estimated from graphs of K_{ex} versus time-of-day (Evans, 1957a, pp. 81-84).

b. Operating time at power levels greater than 10 MW taken from graphs of temperature and power versus time (Evans, 1957a, pp. 61-76).

c. Operating time at power levels greater than 10 MW assumed to be 4 hours per run based on general test practice (Evans, 1957a, p. 56).

d. The activation product release was assumed to be 5 Ci/h multiplied by the number of hours the reactor operated at power levels exceeding 10 MW.

e. The noble gas release was assumed to be 2 Ci/h (decayed to 10 minutes) multiplied by the number of hours the reactor operated at power levels exceeding 10 MW.

f. The Ar-41 release was estimated assuming a production rate of 2.8 Ci/MW-h and a reactor power of 11.8 MW for the number of hours the reactor operated at power levels exceeding 10 MW.

based on observed release rates (Evans, 1957a, p. 85), was assumed during the times that the reactor operated at power levels in excess of 10 MW. No release was assumed for operations at power levels below 10 MW. These assumptions result in an additional release of 90 Ci of activation products and a total estimated release during IET #8 of 243 Ci of particulate activity. Table A-28 provides a summary of measured and assumed releases of particulate activation products during IET #8 operations.

Screening calculations were performed using the RSAC-4 computer code and internal dose rate conversion factors (DOE, 1988b). These calculations showed that Mo-99 produced a larger radiological doses per curie of release than any of the other activation products of molybdenum or manganese. The conservative assumption was made that the total estimated particulate release of 243 Ci was comprised entirely of Mo-99.

Fission Product Releases--Although the operations report stated that there was no fission fragment evolution during IET #8 (Evans, 1957a, p. 55), the assumption was made that noble gases were released at a constant rate of 2 Ci/h (expressed as 10-minute-decayed curies) during those periods when the reactor operated above 10 MW. No release of noble gases was assumed to occur at lower power levels. The release rate of 2 Ci/h was chosen because it represents one of the lower rates of release reported during IETs #3, #4, and #6. Because only noble gases were assumed to be released, a complex operational history was not modeled using the RSAC-4 computer code. This is justified because most of the noble gases have relatively short half-lives and come into equilibrium in the reactor core within several hours of the start of a test run. The RSAC-4 computer code was used to calculate the inventory of radioactive material in the reactor after 6 hours of operations at a reactor power level of 11.8 MW. The 6-hour time period, rather than 4 hours, allows more time for the ingrowth of noble gases in the reactor core. The noble gas release fraction was chosen through an iterative process so that the appropriate total release of noble gases was obtained for each day of operation. A total of 109 Ci of radioactive noble gases was assumed to be released during IET #8. A summary of the assumed airborne fission product releases during IET #8 is provided in Table A-28.

Uranium Releases--Based on the information in Evans (1957a) that there was no fission fragment evolution during the test and no reported damage to the metallic fuel; no uranium was assumed to be released during IET #8.

Argon-41 Releases--A total of 1800 Ci of Ar-41 was estimated to have been released during IET #8, with 1370 Ci remaining after accounting for radioactive decay during transport from IET to the INEL Site boundary. The release was estimated based on a production rate of 2.8 Ci/MW-h and an assumed power level of 11.8 MW during the time period the reactor was operated at power levels exceeding 10 MW. No releases of Ar-41 were estimated for operations below a power level of 10 MW.

IET #9 (September 1957)

IET #9 was a series of critical experiments performed to determine the nuclear characteristics of the HTRE No. 2 reactor and proposed test insert mockups. The types of experiments were similar to those described for IET #7. The exact dates of IET #9 are not available; however, reference is made to their performance in September 1957. This date is reasonable because it falls within the time period after the completion of IET #8 on August 28, 1957, and the beginning of IET #10 on December 12, 1957.

Several references to IET #9 were made in operations reports for other tests. A discussion of the results of critical experiments performed in support of IET #11 states:

> The critical experiments for the 1-C insert were performed during IET-9 (September 1957) using the insert 1-A condiguration [sic] ... and were performed for the purpose of evaluating a modification to the insert which involved the replacement of the center fuel cartridge of the 1-A with a bar of hydrided zirconium... (Evans, 1958b, p. 11).

Reference was also made to the IET #9 critical experiments in a report describing IET #12 (BOOT):

> As a means of detecting the loss of fuel caused by the meltdown it was requested that Kex values be obtained before and after the test [IET #12]. Since it was determined during IET #9 that an insert fuel cartridge in tube 6 was worth approximately 1.24% ΔK/K, the loss in Kex could be a measure of the extent of damage or loss of fuel due to the meltdown (Devens et al., 1958, p.13).

Reference is made in Flagella (1962, p. 35) and Evans (1960b, p. 29) to the critical experiments performed on the insert 2-A mockup of the actual insert 2-B used in the power tests of IET #10. Neither report provides a specific date for the experiments. These critical experiments were performed using insert 2-A and insert 2-A-modified. The insert 2-A was composed of uranium fuel sheets and beryllium metal. To perform more detailed checks of the power distribution, two sections of the mockup were loaded with ceramic fuel tubes that would be used in the insert 2-B and BeO moderator slabs (Evans, 1958a, pp. 20-45). The discussion of results of critical experiments includes Figure 1.9.2, "IET 9-2-A Modified, Temp - 70°, Radial Power Distribution Made with Clad Uranium Wire" (Evans, 1958a, p.36).

As previously described for IET #7, the critical experiments were performed with nuclear mockups of the test inserts. These mockups were not designed for power operations in the reactor; rather, they were operated at very low power levels. The total power generated during a critical experiment normalizing run for insert 2-A was 4.7 W (Evans, 1958a, p. 45). This power level was more than a million times smaller than those encountered during the power testing of the insert 2-B. No radioactive emissions were assumed to have occurred during the low-power critical experiments of IET #9.

<u>IET #10 (December 20, 1957 through March 6, 1958)</u>

IET #10 was performed between December 12, 1957, and March 6, 1958. Releases of airborne radioactivity were documented to have occurred between December 20, 1957 and March 6, 1958. This test series involved the evaluation of insert 2-B.

> The primary purpose of the insert 2B experiment was to operate the HTRE No. 2 reactor system at powers and air flows sufficient to produce a maximum fuel-element temperature of 2750°F, and in a

> post operational examination to evaluate the insert 2B materials for physical and mechanical stability.
>
> The insert 2B test was the first in a series of ceramic tests. Very little was known, at the time of this test, about the operational characteristics of ceramics. The manufacturing and design analysis were likewise in early stages of development.
>
> Some difficulty was encountered in obtaining the design temperature in the insert at the specified power level. During the power operations, parent-core fuel-cartridge temperature limits were reached on the initial attempts to obtain insert design conditions. As a consequence, the HTRE No. 2 had to be returned to the hot shop on [on two occasions] for adjustment of flow to the insert, by plugging holes in the front orifice plate. Later, the insert was operated at a temperature of 2830°F maximum for the first time with an exit air temperature of 2300°F. The insert was operated at these design conditions for periods of time totalling 100 hours, after which the insert was returned to the hot shop for disassembly and examination of the insert (Flagella, 1962, p. 75).

During the disassembly and inspection of the insert following IET #10, some of the fuel tubes in the tenth stage were white in appearance, indicating that the UO_2 fuel had been released from the BeO ceramic matrix as a result of operations at temperatures of 3000 to 3200°F (Flagella, 1962, pp. 75-76).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history and insert fission rates recorded by Foster et al. (1958, pp. 30-31). Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded insert fission rates for each run of the test series for the time period indicated by the start and stop times. The insert was assumed to generated about 7.4% of the total reactor power (Foster et al., 1958, p. 51).

Airborne Radioactivity Releases. The radioactivity released during IET #10 was assumed to be composed of fission products, uranium, and Ar-41. Releases were evaluated for the winter season (December 20, 1957 through February 25, 1958) and the growing season (March 1 through March 6, 1958). The fission product release was calculated for each of the runs indicated in the test series operational history. Uranium releases were assumed to occur during the reactor operations and were estimated on the basis of measured

beryllium releases. Argon-41 was produced in proportion to the energy produced by the reactor during each period of operations.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 1.4E-4 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run for the winter season and the growing season. This value, when multiplied by the test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 1000:1. Several possible values for this ratio were considered. This ratio produced results 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. In an iterative process, the values for the noble gas and radioactive solids release fractions were then selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product release rates for some test runs based on spot sampling data are summarized for several runs in Foster et al. (1958, p. 44) and are reported as 17-second decayed curies. A release rate of 5130 Ci/h was assumed for test runs before February 8, and a release rate of 7500 Ci/h was assumed for February 15 and 16. On March 6, the release rate was assumed to be 31,400 Ci/h. The total fission product release for IET #10, expressed as 17-second decayed curies, was estimated to be 7.17E+5 Ci during the winter season and 9.33 E+5 Ci during the growing season.

Uranium Releases--The estimated uranium release during IET #10 was based on measured average release rates of beryllium in the airborne effluent (Foster, 1958, pp. 27-28). Because the fuel tubes were comprised of a mixture

of beryllium and uranium, the abrasion of material from the walls of the fuel tubes would release uranium in proportion to the amount of beryllium released. A total of 4.82 g of 93.4% enriched uranium was estimated to be released during the winter season and 1.94 g during the growing season.

Argon-41 Releases--A total of 4000 Ci of Ar-41 was estimated to have been released during the winter season and 1310 Ci during the growing season. The production rate of Ar-41 was assumed to be 2.8 Ci/MW-h.

IET #11 (March 20 through April 14, 1958)

IET #11 was performed between March 12 and April 14, 1958. Releases of airborne radioactivity were documented to have occurred between March 20 and April 14, 1958, when the reactor was operated at power levels exceeding 120 kW. This test series involved the evaluation of the insert 1-C. The metallic fuel used in the insert was of the concentric ring design.

> Insert 1C was designed for a reactor experiment to evaluate unclad, slotted, hydrided zirconium as a core neutron-moderating material.
>
> The test program called for the gradual increase in reactor power and temperature until design conditions were reached. When the reactor power was increased to produce a 700°F insert moderator temperature, the stack gas radiation monitoring equipment recorded fresh fission products in the effluent, with a level of discharge of 700 curies per hour (measured 17 seconds after release) or 14 curies per hour (measured 10 minutes after release).
>
> This unexpected release was probably the result of uranium oxide, deposited in the lower cocoon during the insert 2B operation, fissioning in the neutron-flux field below the reactor core. To assure that a fuel cartridge had not ruptured, the CTF was returned to the hot shop for examination and cleaning. In the hot shop, the cocoon was flushed with nitric acid and water, removing a total of 8.4 grams of U^{235}. It was established that no fuel-cartridge rupture had occurred.
>
> When testing was resumed the reactor power level was increased to 11 megawatts to produce the 1200°F moderator temperature. At this power level, the release was measured at 380 curies per hour (measured 17 seconds after release) or 7.7 curies per hour (measured 10 minutes after release) (Flagella, 1962, p. 76).

> The reactor was subsequently operated in intervals summing to 100.24 hours ... During the operation, the reactor was shutdown five times due to unfavorable wind direction and three times due to calculated lung dose (calculations based on stored fission products, which if released, would produce a lung dose exceeding 0.5 rads to an offsite location). The test was successfully completed on April 14, 1958 (Evans, 1958b, p. 28).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Evans (1958b, p. 29). Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the maximum power level recorded for each run of the test series. The operating time was estimated by dividing the recorded energy release for the run by the maximum power during the run.

Airborne Radioactivity Releases. The radioactivity released during IET #11 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 120 kW. Uranium releases were assumed to occur during the reactor operations and were estimated on the basis of uranium release rates measured during IET #14. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Releases--No detailed information concerning daily releases of fission products was available. Based on the summary information contained in the operations report (Evans, 1958b, pp. 27-28), the total fission product release rate was assumed to be 14 Ci/h during the first 3 days of operation, before the uranium was cleaned from the reactor cocoon. A release rate of 7.7 Ci/h was assumed for the remainder of the operations. The operating time was estimated by dividing the recorded energy release for individual test runs by the maximum power achieved during the run. The total release, expressed as 10-minute-decayed curies, was estimated to be 965 Ci.

The fission product releases appear to arise from the unclad fuel debris contained in the reactor cocoon as a result of IET #10 operations. Therefore, release fractions used in the RSAC-4 computer code were assumed to be in the same proportions as those indicated for a prolonged melt condition reported in

Baker et al. (1959, p. 10). Under this assumption, the release fractions for noble gases and halogens were equal, the release fraction for radioactive cesium was a factor of 48 lower than the noble gas release fraction, and the release fraction for other radioactive solids was 7700 times lower than the noble gas release fraction. In an iterative process, the values for the noble gas and radioactive solids release fractions were then selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for the individual runs.

Uranium Releases--Measurements of airborne uranium releases during IET #14 indicated a maximum release rate of about 1 μg/s (Pincock, 1959, pp. 114-118). During the first 3 days of operation of IET #11, before the uranium was cleaned from the reactor cocoon, the uranium release rate was assumed to be 10 times the rate observed during IET #14. For the remaining portion of the test, the uranium release rate was assumed to be 1 μg/s. A total of 0.714 g of 93.4% enriched uranium was estimated to be released.

Argon-41 Releases--A total of 3670 Ci of Ar-41 was estimated to have been released during IET #11 on the basis of a production rate of 2.8 Ci/MW-h.

IET #12 (BOOT) (May 2, 1958)

IET #12, also referred to as Burnout One Tube (BOOT), was conducted from April 21 through May 7, 1958. Releases of airborne radioactivity occurred when the reactor operated at power levels exceeding 110 kW on April 30, May 2, and May 6, 1958. This experiment involved the intentional melting of one of the metallic fuel cartridges in the insert 1-D.

> The purpose of the test was to evaluate the hazard of a "one-tube melt" in a full size reactor similar in design and containing the fuel element and moderator materials of insert 1D (Flagella, 1962, p. 78).
>
> Insert 1D was similar to inserts 1B and 1C in that it consisted of seven hexagonal hydrided-zirconium blocks. The six outer blocks contained similar fuel cartridges of the metallic concentric ring design. The center block contained a beryllium bar for the

> purpose of boosting the flux in the center of the insert and reducing the flux gradient across the diameter of the six outer blocks. Two of the outer blocks [2_i and 6_i] contained remotely operable pneumatic valves at the inlet end, for the purpose of reducing the airflow to the fuel elements during operation (Flagella, 1962, pp. 77-78).
>
> A controlled amount of fission-product activity was required for this experiment. The reactor was operated at power [on April 30, 1958], not exceeding 1750°F on the insert fuel cartridge, until approximately 23 megawatt hours had been accumulated. The resulting fission product activity was then permitted to decay for approximately 48 hours. During the rise to power on this fission-product storage run, the airflow restrictor valves were operated in the opened and closed positions, and the temperature data were recorded.
>
> Following the decay period of approximately 48 hours, all CTF, exhaust duct, and field sampling devices, as well as weather instruments, were made ready for the meltdown [test].
>
> The reactor was again taken to a power level which produced a maximum fuel-cartridge temperature of 1750°F on tube 6_i. After a period of 5 minutes, the airflow restrictor valve on tube 6_i was closed [at 5:15 p.m. on May 2, 1958], causing the fuel cartridge to overheat and melt. This phase of the test ended with an automatic reactor scram resulting from short periods of increased reactivity recorded by the fission chambers (Flagella, 1962, p. 78).

The May 2 test was referred to as BOOT 1 by Devens et al. (1958, p. 8).

On May 6, an attempt was made to melt the fuel in cartridge 2_i. This phase of the operation was referred to as BOOT 2 by Devens et al. (1958, p. 8).

> After cartridge 6_i was melted, consideration was given to the thought of melting cartridge 2_i as it was already instrumented and had an independent valve system. The valve used to restrict the air to cartridge 6_i was opened on May 6, 1958, and the reactor was taken stepwise to 1, 5, 20, and 40 percent of full power. Thermocouples still in working order on cartridge 6_i indicated temperature rises comparable to those noted when the valve restricting air to tube 6_i was closed. Detection instruments also indicated increased activity in the effluent. It was concluded that a restriction somewhere in tube 6_i was limiting the coolant so severely that any additional power would cause cartridge 6_i to remelt, thereby nullifying any chances of gaining information from either the first melt or the proposed second one. The decision

was then made to return the reactor to the hot shop for dismantling and examination (Baker et al., 1959, p. 52).

Fission Product Inventory. The inventory of radioactive fission products in the fuel cartridge of tube 6_i was based on the reactor operating history and the fraction of the reactor power produced in the 6_i tube. The reactor operating history was recorded in Devens et al. (1958, p. 34). Inventory calculations were performed with the RSAC-4 computer code using actual reactor power levels and operating times. The total power generated by the insert was 11% of the total reactor power, with individual fuel cartridges in the insert tubes generating between 1.5 and 2% of the total reactor power (Baker et al., 1959, p. 27). The assumption was made that the fuel cartridge in tube 6_i produced 2% of the total reactor power.

Airborne Radioactivity Releases. The airborne radioactivity released during this test was assumed to be composed of fission products, uranium, and Ar-41. The majority of the fission product release occurred on May 2, with the remainder released from the reactor operations on May 6. There was no significant fission product release during the April 30 operation. Uranium was released at the time of the fuel melt on May 2 and again during the operations on May 6. The effluent monitoring data were not adequate for determining the amount of uranium released on the specific dates; therefore, only an estimate of the total uranium released during the course of the test is provided. Argon-41 was produced in proportion to the energy produced by the reactor. Because the radioactive airborne releases on April 30 and May 6 were less than the releases on May 2, they were added to those estimated for May 2 and assumed to have been released on that date.

Fission Product Releases--The fission product release during the May 2 test was estimated using information collected during the posttest examination of the insert 1-D. Examination of the insert indicated that 50% of the fuel in the tube 6_i was damaged, which indicates that at most, about 50% of the fission products contained in this cartridge was released from the fuel. During the examination, the slag and individual stages of the fuel cartridge in tube 6_i were removed and weighed.

> The original gross weight of the last nine stages was approximately 2000 grams. Since only about 1157 grams could be accounted for, about 843 grams (approximately four stages) could not be accounted for and are assumed to have passed along to or through the exhaust system (Baker et al., 1959, p. 106).

Thus, about 42% of the mass from the fuel cartridge in tube 6_i could not be accounted for. The release fractions assumed in the RSAC-4 computer code for the May 2 test were assumed to be 50% for the noble gases, 42% for the halogens, and 42% for the radioactive solids in the fuel cartridge from tube 6_i. These assumptions result in a total airborne release of radioactivity of 2.9E+4 Ci, expressed as 10-minute-decayed curies, calculated with the RSAC-4 computer code. This release bounds the total fission product release indicated by the radiation monitor on the 76-in. effluent duct of 2.1E+4 Ci and the stack monitor indication of 1.3E+4 Ci (Devens et al., 1958, pp. 15-16).

The airborne release of fission product activity from the May 6 operation was conservatively estimated based on effluent sampling results. The largest average percent release for I-131, I-133, I-134, and I-135 measured with the charcoal trap samplers in the 76-in. effluent duct and the hot duct was reported to be 0.55% (Devens et al., 1958, p. 17). This average was calculated assuming that the damaged fuel cartridge in tube 6_i was still generating power at the same rate as before the meltdown. On the basis of the above information, the release fraction used in the RSAC-4 computer code for calculating airborne emissions of halogens was 0.0055 for the May 6 operations. The release fraction for noble gases was assumed to be equal to that determined for the halogens. The release fraction for radioactive solids was assumed to be in the same relative proportion to the halogen release fraction as the release fractions for strontium and iodines listed for a "prolonged melt" condition in Table 1 of APEX-445 (Baker, 1959 et al., p. 10). This results in a 7.14E-7 release fraction for radioactive solids. The total fission product release on May 2, expressed as 10-minute-decayed curies, was estimated to be 1.59E+3 Ci.

Uranium Releases--For consistency with the fission product releases, the fraction of uranium released during the IET #12 testing was

assumed to be 42%. This corresponds to a release of 175 g of 93.4% enriched uranium and represents the total uranium released on May 2 and 6. Effluent monitoring data were not adequate for determining the proportion of the release attributable to individual dates.

Argon-41 Releases--A total of 70.0 Ci of Ar-41 was estimated to have been released during reactor operations on April 30 and May 2, 1958, based upon a production rate of 2.8 Ci/MW-h. An additional 4.6 Ci of Ar-41 was produced on May 6. After accounting for radioactive decay during transport, 65.2 Ci of Ar-41 remained at the INEL Site boundary.

IET #13 (HTRE No. 3 Excursion) (November 18, 1958)

The critical experiments and low-power testing phase of the HTRE No. 3 operations was designated IET #13. IET #13 was performed between September 8 and November 18, 1958. It ended when a nuclear excursion occurred. Only the nuclear excursion of November 18, 1958, was responsible for a release of radioactive material to the environment. The critical experiments and low-power testing did not produce any significant airborne releases when compared with the nuclear excursion.

> The initial low-power testing of the D102A reactor was conducted at the Low Power Test (LPT) Facility during the months of April, May, and June 1958. These tests were conducted to confirm design parameters and to provide basic nuclear data on the as-built system prior to high-temperature operation at the Initial Engine Test Facility (IET). The first series of tests included initial loading, reactivity measurements, power mapping, power profile alterations, and elevated-temperature power mapping. Essentially all test objectives were attained and the reactor was found to conform with the design parameters within the specified tolerances. The reactor poison configuration was altered four times during this series to achieve a radial power profile calculated to be flat within ± 7 percent across the core. The maximum fuel plate local-to-average core power scallop was 1.14.
>
> On September 8, 1958, the HTRE No. 3 was moved to the IET and preparations were started for Phase 1 testing. Since this test series was to provide a shakedown of the power plant, the first operations were a series of tests of the service air system to insure that it would supply adequate cooling to various components on the engines and dolly during reactor operation. Next, the increase in reactivity resulting from draining the front, side,

> and rear shield compartments was determined. The increase was found to be 3 percent $\Delta k/k$. The worth of shim rod 243 was measured as 0.156 percent $\Delta k/k$ as compared to 0.161 percent $\Delta k/k$ at LPT. The excess reactivity was determined to be 1.88 percent $\Delta k/k$ with the shield full of water.
>
> Reactor power was determined by exposing foils attached to dummy control rods placed in the active core. Foil count rate was corrected to saturated activity and then compared to the LPT dummy rod foil activity for which the core power had been determined by a detailed flux map. A correlation was also obtained in order to calibrate the nuclear instruments.
>
> The moderator and control rod heating rates were measured with Bragg-Gray ion chambers. The maximum heating rate measured was 0.0755 watts/g-mw [sic].
>
> Rather extensive gamma ray measurements were made external to the shield by suspending pocket dosimeters and film packs from the test cell ceiling. This system of measurement worked fairly well and showed a reproducibility with a range of 20 to 30 percent.
>
> On November 18, 1958, a power excursion occurred, the result of the dynamic and shim rods being withdrawn by the control system under the influence of an erroneous reactor power indication. The false power indication was due to a high-series-resistance noise filter in the output of each ion chamber power supply. Another contributing factor was an improper voltage setting for these power supplies. The mechanism of reactor shutdown was partially self-initiated. It appears that a scram due to fuel element temperature indication (or due to melting of the thermocouple lead wires) and a reactivity loss of about 2 percent caused by melting and collapse of fuel rings occurred within a very short time interval and that both contributed to reactor shutdown. Following this excursion, the power plant was returned to the Hot Shop for inspection and replacement of the fuel elements (Linn et al., 1962, p. 39).

The stack monitor was operating and recorded the release to the environment (GE, 1959, p. 74).

> The IET stack monitor samples the stack effluent at the 80-foot level. The sample is collected on a moving filter-paper tape, which passes over a scintillator head and produces a record of the collected activity. The stack monitor ran continuously during the operations at the IET and recorded the entire event.

The activity release resulting from the incident is described in GE (1959, p. 10):

> Activity was released from the exhaust stack, and a narrow band of fallout occurred that was contained fully within the boundaries of the National Reactor Testing Station. The maximum dose rate observed in the Assembly and Maintenance area and approximately 3000 feet from the cloud centerline was 0.04 milliroentgen per hour. At a distance of 3 to 5 miles from the IET, the measured fallout was 1.25 microcuries per square meter for I^{135}. The ratio of I^{131} to I^{135}, obtained from vegetation, was 0.0089. The maximum fallout observed, at about 4 hours after the incident, measured 0.8 to 2 milliroentgens per hour at contact roughly 1-1/2 miles from the IET.

An examination of the reactor core was performed in the Hot Shop following the nuclear excursion.

> Examination of the reactor indicates that all of the fuel cartridges experienced melting in the middle stages. The amount of heat required to produce such melting is consistent with the total energy release of 770 megawatt-seconds as measured by an indium foil attached to the reactor (GE, 1959, pp. 7-8).

Fission Product Inventory. The fission product inventory present in the core at the end of the IET #13 low-power testing, just before the excursion, was based on the operating history recorded in Devens (no date, pp. 59-60). The nuclear excursion, occurring at the end of the low-power testing, was modeled as a 10-second reactor operation at a power level of 77 MW, consistent with the estimated energy release of 770 MW-s. The inventory was modeled by the RSAC-4 computer code preserving the time-history of the low-power testing and the nuclear excursion.

Airborne Radioactivity Releases. Fission-product, uranium, and Ar-41 releases were considered for modeling the airborne release of radioactivity resulting from the reactor operations based on the fission product inventory present in the core, the amount of core damage, and the resulting energy release during the nuclear excursion.

Fission Product Releases--A number of factors govern the extent of the fission product release resulting from the power excursion. The previous operating history of the core had generated a small amount of longer-lived radionuclides within the core. The 770 MW-s nuclear excursion produced a

substantial amount of relatively short half-lived activity, and the uncontrolled heating of the core caused fuel melting and significant damage to the fuel assemblies. Another factor important to the amount of activity released to the environment was the amount of core cooling at the time of the excursion.

> The airflow for this experiment was provided by two electrically driven blowers supplying approximately 3 pounds of air per second, passing the air through the reactor and out the No. 2 jet engine turbine. The jet engine motored at approximately 600 rpm, and this indication was evident to the operator at the main console at all times. The airflow was adequate for this condition. There was no manipulation of any duct valves throughout the operation (GE, 1959, p. 9).

The cooling airflow through the reactor was about 2% of normal, but it was sufficient, according to the operations personnel, for the conditions of the test. This flow rate amounts to a linear flow rate of about 17 in./s in the 76-in. duct and would allow considerable settling of damaged fuel particles within the duct. To conservatively estimate the release, settling that would have occurred within the 76-in. duct has not been considered in the release calculations.

The iodine release fraction was conservatively estimated to be 2.44% of the inventory present in the core at the end of the power excursion. This value was calculated by dividing the reported iodine release for each of the iodine isotopes by the respective RSAC-4 calculated 10-minute-decayed curie inventory present in the core following the excursion. The resulting release fractions were more consistent than, and also bounded, reported release fractions (Wilks, 1959, p. 14). The largest estimated release fraction was 0.0244 for I-131. This value was rounded to 0.025 and multiplied by 1.5 to ensure a conservative release fraction of 0.0375 for halogens, and it was used in the RSAC-4 computer code.

Because of the similarity of the nuclear excursion to the "quick melt" experiment performed at the Oak Ridge Laboratory before the BOOT test (IET #12), the amounts of noble gases and solids released during the nuclear excursion were based on the "quick melt" release fractions (Baker et al.,

1959, p. 10). Data show the noble gas release fraction to be 0.094, 2.5 times the iodine release fraction. Because there was so much fuel damage and the core cooling airflow was limited, the release fraction of the cesiums was assumed to be 1/15 of the noble gases rather than the 1/20 value found in Baker et al. (1959). For the less volatile radioactive solids, the relative release fraction was assumed to be 1/250 of the noble gases [i.e., a value less than for the cesiums but considerably higher than reported for the strontium isotopes in Baker et al. (1959)]. Therefore, for the purposes of this analysis, it was assumed that 0.0375 of the core iodines, 0.094 of the noble gases, 0.0063 of the cesiums, and 0.000375 for the remaining core solids were released. These release fractions result in a total release of 9730 10-minute-decayed curies, in contrast to the 430 Ci documented in Wilkes (1959, p. 14).

Uranium Releases--The uranium release during the nuclear excursion was calculated using the same release fraction (0.000375) assumed for the radioactive fission product solids other than cesiums. A total of 72 g of 93.4% enriched uranium was estimated to be released.

Argon-41 Release--A total of 0.6 Ci of Ar-41 was estimated to have been released during the nuclear excursion on the basis of a production rate of 2.8 Ci/MW-h and an energy release of 770 MW-s.

IET #14 (April 24 through May 19, 1959)

IET #14 was performed between March 27 and May 20, 1959. Releases of airborne radioactivity were documented to have occurred between April 24 and May 19, 1959, when the reactor was operated at power levels exceeding 100 kW. Small releases of fission product activity were also documented to have occurred between April 17 and 23, 1959, when the reactor operated at power levels below 100 kW. This test series involved the evaluation of the L2A-1 insert cartridge. The insert cartridge contained both fueled and unfueled BeO ceramic tubes. There was no coating on the inside surfaces of the fueled tubes (Pincock, 1959, p. 11).

A total of 100.25 hours was accumulated on the insert fuel cartridge at a maximum insert fuel temperature of approximately 2500°F.

The objectives of the test were:

1. To evaluate the operational effect of water vapor corrosion on fueled BeO tubes operating at a constant reactor mixed mean discharge air temperature over a period of approximately 100 hours.

2. To measure the fission product release rate from uncoated fueled tubes as a function of temperature and operating time (Pincock, 1959, p. 10).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1959, p. 17-19) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a power level greater than 100 kW. The insert was assumed to generate 2.7% of the total reactor power (Pincock, 1959, p. 71).

Airborne Radioactivity Releases. The radioactivity released during IET #14 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 100 kW. The small releases of fission product activity (<1 Ci) that occurred from April 17 through 23, when the reactor was operating at power levels of less than 100 kW, were not individually modeled. Uranium releases were estimated based on the results from effluent samples analyzed for uranium. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Releases--The inventory-normalized release rate for radioisotopes of iodine was assumed to be the same as the value determined for IET #10 and was equal to 1.4E-4 Ci/h per curie of the isotope of iodine present in the insert inventory at the end of a test run. This value, when multiplied by the test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The

ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 1000:1. This ratio was chosen because uncoated fuel tubes were used in the insert, and it is the same ratio assumed for IET #10. In an iterative process, the values for the noble gas and radioactive solids release fractions were selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product releases for test runs based on spot sampling data are summarized in Pincock (1959, p. 120) and are reported as 10-minute-decayed curies. The total fission product release for IET #14 was reported as 8565 Ci. An error was made in the summation of the tabulated releases in the operations report; the correct total fission product release was 8656 Ci.

Uranium Releases--The uranium release during IET #14 was estimated from the analysis of five charcoal trap samples for uranium. The analyses indicated that in all cases the uranium content was at or below the limit of detection.

> The rate of release of U corresponding to the limit of detection would be about 1 micro-gram per second. If this release rate is assumed a total release during the 100 hours would be about .36 grams of uranium (Pincock, 1959, pp. 114-118).

To obtain a conservative estimate of the potential release of uranium, the release rate of 1 μg/s was applied to the entire time period that the reactor was operated in excess of 100 kW, rather than just the 100 hours of time at the specified test conditions. A total of 0.476 g of 93.4% enriched uranium was estimated to be released.

Argon-41 Releases--A total of 3800 Ci of Ar-41 was estimated to have been released during IET #14 on the basis of a production rate of 2.8 Ci/MW-h.

IET #15 (June 3 through June 24, 1959)

IET #15 was performed between May 27 and June 24, 1959. Releases of airborne radioactivity were documented to have occurred between June 3 and June 24, 1959, when the reactor was operated at power levels exceeding 100 kW. This test series involved the evaluation of the L2C-1 insert cartridge.

> The L2C-1 fuel cartridge was of concentric ring design, with fuel sheet made of a chromium-UO_2-titanium core clad with an iron-chromium-yttrium alloy (Evans, 1959, p. 6).
>
> From this operation data would be obtained to evaluate:
>
> 1) Endurance capabilities of the advanced metals at a design temperature of 2000°F for extended periods of time (planned endurance testing to sum 120 hours or more).
>
> 2) The structural and metallurgical integrity of the fuel sheet in this particular cartridge design.
>
> 3) The nature and extent of fuel sheet damage, if any, and the effect on cartridge performance.
>
> 4) The performance potentialities of the cartridge.
>
> The operation was successfully conducted to accumulate 80.75 hours at an insert extrapolated fuel sheet temperature of 2015°F ... The operation was terminated after 80.75 hours due to a release of fission products of such a quantity as to indicate fuel sheet rupture of an extent sufficient to warrent [sic] inspection (Evans, 1959, p. 6).

The insert was visually examined after the completion of the IET #15 testing. No damage had occurred to the outer fuel sheet of the cartridge; however, fuel sheet blisters were observed on the inner fuel sheets. In some instances the blisters had ruptured (Evans, 1959, p. 42).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Evans (1959, p. 33) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a

power level greater than 100 kW. The insert was assumed to generate 4.9% of the total reactor power (Flagella, 1962, p. 82).

Airborne Radioactivity Releases. The radioactivity released during IET #15 was assumed to be composed of fission products, uranium, and Ar-41. The releases were assumed to occur during two time periods to account for the change in the relative amounts of fission product activity released before and after fuel blistering occurred. The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 100 kW. Uranium releases were estimated based on a conservative evaluation of the extent of the observed fuel blistering. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). Effluent monitoring results for radioactive isotopes of iodine were summarized in Evans (1959, p. 24).

> After thirty hours the iodine release rate recorded from samples taken from the 80-foot stack level increased rapidly with other effluent sampling devices showing small releases. At this time it was more or less apparent that some release of fission products from the fuel sheet was occurring.

Based on a review of the effluent monitoring data for radioactive isotopes of iodine, the airborne releases were considered in two periods: (1) June 3 to 15, before the development of significant fuel sheet blisters and (2) June 16 to 24, when the effects of blistering were clearly observed. Inventory-normalized release rates for I-131, I-133, and I-135 were calculated on the basis of the analysis of charcoal trap samples from the 80-ft level of the stack. From June 3 to 15, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 4.15E-6 Ci/h of I-135 released per curie of I-135 inventory in the insert at the end of a test run. From June 16 to June 24, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 4.62E-5 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test

run. These values, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases.

The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 2000:1. In an iterative process, the values for the noble gas and radioactive solids release fractions were selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product releases for test runs based on spot sampling data are summarized in Evans (1959, p. 41) and are reported as 10-minute-decayed curies. The total fission product release IET #15 was reported as 415 Ci during the first period of operations from June 3 to 15 and 484 Ci from June 16 to 24.

Uranium Releases--The estimated uranium release during IET #15 was based on the observation that blisters had formed on the fuel sheets and that some of the blisters had ruptured. The damage appeared to be comparable to that observed in IET #6, where 5 g of uranium was assumed to be released. The IET #15 release was assumed to be no larger than 5 g of 93.4% enriched uranium. Ten percent of the uranium was assumed to be released during the first period of operations from June 3 to 15. The remaining 90% was assumed to be released during the second period of operations from June 16 to 24.

Argon-41 Releases--A total of 1730 Ci of Ar-41 was estimated to have been released from IET #15 during the first period of operations from June 3 to 15, and 995 Ci was estimated to have been released during the second period from June 16 to 24. The Ar-41 production rate was assumed to be 2.8 Ci/MW-h.

IET #16 (October 9, 1959)

This series of tests, conducted between July 28 and October 9, 1959, was the first power test of the HTRE No. 3 and was conducted to determine the operating characteristics of the horizontal core. Releases of airborne

radioactivity were modeled as if they had all occurred on October 9, 1959. The measurements obtained were (a) temperature profiles and temperature measurements of components including the fuel elements, the reactor shield, and the exhaust ducting; (b) control and shim rod calibrations and excess reactivity measurements; (c) power calibrations using foil measurements; (d) comparison of thermocouple and cooling-air flow measurements taken to measure reactor power; (e) chemical engine performance measurements; (f) engine shutdown characteristics; (g) measurements of nuclear heating caused in various engine components; and (h) isodose measurements and isodose profiles caused by gamma radiation and thermal and fast neutrons. One important aspect of the isodose profiles was the crew cavity shield, located 75 ft from the center of the core. Detailed measurements of the shielded cavity were performed. According to Showalter (1959, p. 65):

> The Test Program required a complete data scan and system data for each steady state point in the power test series. These data plus the data from the additional check runs have been sent to Evendale. All of the runs requested were not performed because the IET rupture detection system was not adequate for this power plant. Reactor operation was limited to 10 mw [sic] and/or 1350°F. (hottest reliable fuel element temperature reading) because of this inadequacy.

The IET #16 operation is briefly summarized in Miller et al. (1960, p. 4).

> A total of 95 megawatt hours was accumulated. Maximum power was 10 megawatts at 52 lbs/sec airflow with maximum fuel element temperature of 1350°F and reactor exit air of 1025°F.

Although 95 MW-h is a relatively small power generation (equivalent to 9.5 hours at the maximum power of 10 MW), most of the testing was performed at low-power levels over a time span of just 1 month.

Fission Product Inventory. Because the radioactive material release from this test was relatively small compared to the other IET series, the radiological effect to an offsite individual was maximized by assuming that the operation occurred during one 9.5-hour period where the reactor was

operated at the maximum allowed power level of 10 MW. The release to the environment was assumed to occur during this same 9.5-hour period. Analysis of the activity during the test series gave indications that the activity released had aged components and had originated from fuel released to the cocoon during the nuclear excursion of November 18, 1958.

Airborne Radioactivity Releases. Radioactivity released during this test series was modeled to consist of fission products, uranium from the cocoon of the reactor, and Ar-41 that was generated during the operation of the reactor.

Fission Product Releases--Release fractions of the fission products for this analysis are treated according to the account described in Showalter (1959, p. 74).

> The background trap samples and all traps collected at lower powers showed traces of old fission products in the gross gamma spectrum.
>
> The set of carbon trap samples collected during Run 29 had sufficient activity that an I^{135} determination could be made.
>
> The six carbon traps sampling from the probes located downstream of the six core sectors had I^{135} activities agreeing with each other within a factor of 5. The highest indicated release fraction was 2.9×10^{-7} and the average of all 6 was 1.4×10^{-7}. The release fraction was calculated on the basis of total reactor power but the actual source of the fission products was probably due to small amounts of uranium in the system from the 1958 [IET #13 nuclear excursion] incident.
>
> The I^{135} release fraction indicated by a carbon trap sampling from the 80 foot level of the stack on the same run was about 5×10^{-8} which corresponds to a gross fission product release rate of about 1 curie/hour (age 10 minutes). Spot samples of the effluent indicated a release rate of about 1.5 curies/hr.

Modeling with the RSAC-4 computer code for IET #16 releases involved the preservation of three factors: (1) a burnup of 95 MW-h, (2) a conservative particulate release of 14 Ci (1.5 Ci/h for a total of 9.5 hours), and (3) a conservative release fraction of 3.0E-7 for iodine isotopes, the highest fraction measured during the test. To preserve these values and arrive at release fractions for the other groups of radionuclides, engineering judgment

and preliminary data from a few iterations of the RSAC-4 computer code were used. To meet the stated criteria of 1.5 Ci/h for particulates, the noble gas release fraction was assumed to be 200 times higher than that for the iodines, and the release fraction for the solids was assumed to be 10% of the halogens release fraction. Therefore, the release fractions used in the RSAC-4 computer code were 3.0E-7 for iodine, 6.0E-5 for the noble gases, and 3.0E-8 for the solids.

Uranium Releases--Uranium was assumed to be released at the same rate as the radioactive fission product solids. A total of 0.00579 g of 93.4% enriched uranium was estimated to be released during IET #16.

Argon-41 Releases--During the operation, 266 Ci of Ar-41 were released on the basis of a production rate of 2.8 Ci/MW-h.

IET #17 (November 2 through December 12, 1959)

IET #17 was performed between October 12 and December 12, 1959. Releases of airborne radioactivity were documented to have occurred between November 2 and December 12, 1959, when the reactor was operated at power levels exceeding 100 kW. This test series involved the evaluation of the L2E-1 insert cartridge.

> The L2E-1 insert cartridge was emplemented [sic] to further evaluate in a nuclear reactor the high temperature operations of fueled ceramic tubes. The power operation was conducted in light of insert 2B [IET #10] and insert L2A-1 [IET #14] cartridge operations wherein modifications to the fuel tubes (including an ID coating of alumina) were included to reduce or eliminate BeO hydrolysis and crystal growths (redeposition) as well as fission product releases.
>
> The insert cartridge was designed to operate at a maximum fuel tube indicated temperature of 2515°F at a total reactor power [of] 12 megawatts. The design temperature was achieved for the first time on November 10, 1959, at a total reactor power of 10.5 megawatts. Temperature distributions in the cartridge were essentially as expected. The operation was conducted in several successive power tests to accumulate 105.99 hours at the design maximum temperature. During this time there was essentially no change in test conditions from initial values.

> Throughout the endurance testing the release of fission products was monitored. Data indicated that the fractional releases of iodine 131 and 135 did not vary significantly from initial on test values and in comparison to initial L2A-1 data was [sic] slightly lower.
>
> Samples of gross stack activity showed a reduction of between a factor of 3 and 4 from the L2A-1 operation indicating that the Al_2O_3 coating is apparently effective in reducing or delaying the evolution of some fission products (Evans, 1960a, p. 5).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1960e, pp. 16-18) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a power level greater than 100 kW. The insert was assumed to generate 2.16% of the total reactor power (Evans, 1960a, p. 52).

Airborne Radioactivity Releases. The radioactivity released during IET #17 was assumed to be composed of fission products, uranium, and Ar-41. Releases were evaluated for the growing season (November 2 to 30) and the winter season (December 1 to 12). The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 100 kW. Uranium releases were estimated on the basis of measured beryllium releases. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 2.26E-5 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run for the growing season and winter season. This value, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The

ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 2000:1. This ratio was chosen based on the reported effectiveness of the alumina coating on the fuel tubes in reducing radioactive airborne emissions from the reactor. In an iterative process, the values for the noble gas and radioactive solids release fractions were selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product releases for test runs based on spot sampling data are summarized in Pincock (1960e, pp. 126-127) and are reported as 10-minute-decayed curies. The total fission product release for IET #17 was reported as 1110 Ci during the growing season and 912 Ci during the winter season.

Uranium Releases--The estimated uranium release during IET #17 was based on measured release rates of beryllium in the airborne effluent (Pincock, 1960e, p. 121). Because the fuel tubes were comprised of a mixture of beryllium and uranium, the abrasion of material from the walls of the fuel tubes would release uranium in proportion to the amount of beryllium released. A total of 0.221 g of 93.4% enriched uranium was estimated to be released during the growing season and 0.262 Ci during the winter season.

Argon-41 Releases--A total of 2170 Ci of Ar-41 was estimated to have been released during the growing season and 2010 Ci during the winter season. The production rate of Ar-41 was assumed to be 2.8 Ci/MW-h.

IET #18 (January 6 through February 7, 1960)

IET #18 was designated as the Phase 2 testing of the HTRE No. 3 engine. Testing occurred between December 23, 1959, and February 8, 1960. Releases of radioactive material were modeled to occur between January 6 and February 7, 1960. This test series, which consisted of 126 hours at design conditions, was an extension of the test program outlined for IET #16 to define the operational parameters required to understand the aspects of the reactor, power control, after heat, radiation fields, and shielding. The design power level for this core was 32 MW (Blumberg, 1960, p. 8).

The following is a brief description of the reactor operation:

> The power plant was transported to Initial Engine Test December 14, 1959, for final checkout in the facility prior to testing. The first engine operation was accomplished December 22, 1959, and the first data was taken December 23, 1959, (run No. I-1). The reactor was made critical on December 23, 1959, (run No. I-6).
>
> Damage was sustained to the instrumentation circuitry of the power plant January 6, 1960, as a result of failure of the electrical driven aftercooling blowers. The power plant was returned to the Hot Shop January 7, 1960, for repairs which were completed January 12, 1960. Testing was resumed at Initial Engine Test January 21, 1960, and completed February 7, 1960.
>
> The initial transfer from chemical operation to full nuclear operation was accomplished on January 26, 1960, (run No. 11-12), and the design condition for endurance testing were initially attained at 2358 on 1-26-60 (run 11-32). A total of 126.42 hours of operation was achieved at design conditions with a continuous operation of 64.9 hours at these conditions. Operations were accomplished above 1% of design power for a total of 166.5 hours (Highberg et al., 1960, p. 5).

Fission Product Inventory. The fission product inventory was calculated using the RSAC-4 computer code, reproducing the time periods and power levels at which the reactor operated and was shut down. The reactor operating history was provided in Highberg et al. (1960, pp. 7-12). Following a review of the detailed operation, the history was simplified to correspond with the major periods of operation. This simplified operation is summarized in Table A-29 for the considered periods of operation.

Airborne Radioactivity Releases. As for the analysis of IET #16, the released fission products were assumed to be derived from uranium remaining in the cocoon of the reactor as a result of the nuclear excursion that occurred at the end of IET #13. A small amount of uranium was also released during the test series. Argon-41 was released in proportion to the energy produced by the reactor.

Fission Product Releases--One paragraph of the available GE-ANP literature is devoted to the radioactive effluent from IET #18:

Table A-29. Operational parameters and release quantities for the eight test periods of IET #18

Test Period	Beginning Date of Period	Operation Time (h)	Percent of Total Power	Release 10-Minute-Decayed (Ci)
1	1/06/60	3.3	57	13.5
2	1/23/60	3.4	39	9.5
3	1/26/60	13.0	45	28.9
4	1/26/60	8.6	74	45.2
5	1/27/60	30.38	94	203.4
6	2/03/60	10.66	96	72.8
7	2/03/60	26.33	100	226.4
8	2/05/60	64.8	100	557.3
TOTAL				1157

> Continuous effluent monitoring was maintained to measure and to record the activity released to the atmosphere by the power plant. The maximum output was 8.6 curies/hour (measured 10 minutes after release). The total output for the test series was 1157 curies (measured 10 minutes after release). The maximum release rate for I-131 was approximately 1.5 curies/hour (measured 10 minutes after release). The total offsite inhaled and ingested dose was below measurable amounts during this test series (Highberg et al., 1960, p. 276).

Following the convention used in the majority of the IETs, all fission product radioactivity releases are expressed as 10-minute-decayed curies.

The quote above contains conflicting information; a maximum release rate of 8.6 Ci/h is not consistent with a maximum I-131 release rate of 1.5 Ci/h. An analysis of the IET #18 operation indicated that an I-131 release rate of 1.5 Ci/h would be associated with a total I-131, I-133, and I-135 release rate of at least 14 Ci/h. In all likelihood, the release rate for I-133 and I-135 would be 10 to 100 times that of I-131, which is inconsistent with the maximum 8.6 Ci/h total fission product release rate.

Based on a power level of 10 MW during run 29 of IET #16 and the measured I-135 release fraction of 5E-8 (Showalter, 1959, p. 75), the corresponding I-135 release rate of 3.9E-3 Ci/h was estimated using the

techniques and parameters outlined in Pincock (1959). With the amount of I-135 released being 10 to 100 times that of the I-131, I-135 would be one of the first iodine isotopes detected in the effluent. The associated I-131 release rate would be at most 4E-4 Ci/h at the 10 MW power level during IET #16. During IET #18, the reactor was operated at 35 MW, a power level approaching the design level for this core. Based on the IET #16 release fraction measurement, the I-131 release rate during IET #18 would be projected to be 0.001 to 0.002 Ci/h or about 1.5 mCi/h, rather than the reported 1.5 Ci/h.

As a result of extensive analyses and a review of the operations reports for IETs #13, #16, #18, and #25; all of the HTRE No. 3 tests; and the environmental monitoring data, it was concluded that either (a) the release rate was misstated as 1.5 Ci/h instead of 1.5 mCi/h or (b) the release was inappropriately labeled as I-131 instead of I-135. The more conservative of these assumptions, that the reference was incorrectly made to I-131 rather than to I-135, was used in reconstructing the source term for IET #18.

Not all of the eight operational periods were assumed to have a release rate equal to the maximum of 8.6 Ci/h. This rate was adjusted according to the estimated percent of total power for a given test period, as shown in Table A-29 and to account for the average activity produced over the respective period. The total release for the eight operational periods (expressed as 10-minute-decayed curies) totaled 1157 Ci as shown in Table A-29 and as documented for the test series in Highberg et al. (1960, p. 276).

The average I-135 release rate was assumed to be proportional to the estimated average test period inventory divided by the maximum inventory present at the end of the test period. For example, during the eighth test period, which corresponded with the highest I-135 inventory, the release rate was estimated to be (1.70E+6/1.73E+6) x 1.5 Ci/h or 1.47 Ci/h. The total I-135 release during the 64.8 hour operation of the eighth test period was 95.5 Ci. The RSAC-4 halogen release fraction for this period is 95.5 Ci divided by the I-135 inventory at the end of the period (1.73E+6 Ci) or 5.52E-5. The ratio of the noble gas release fraction to the solids release fraction was assumed to be 2000. The remaining RSAC-4 release fractions for

noble gases and solids were chosen for this period in an iterative process so that the total fission product release amounted to 557.3 Ci (64.8 h x 8.6 Ci/h).

Uranium Release--Uranium releases were not addressed in the operations report for IET #18. To estimate the uranium released, the IET #16 uranium release (0.00579 g) was scaled upwards in proportion to the nuclear energy generated during the test series. Thus, this uranium release was estimated to be 0.33 g of 93.4% enriched uranium.

Argon-41 Release--A total of 13,600 Ci of Ar-41 was estimated to have been released during IET #18 on the basis of a production rate of 2.8 Ci/MW-h.

IET #19 (February 17 through April 30, 1960)

IET #19 was performed between February 9 and April 30, 1960. Releases of airborne radioactivity were documented to have occurred between February 17 and April 30, 1960, when the reactor was operated at power levels exceeding 100 kW. This test series involved the evaluation of the L2E-3 insert cartridge. The cartridge contained hexagonal BeO ceramic tubes, both fueled and unfueled, that were coated on the inside with co-extruded zirconia (ZrO_2) (Pincock, 1960d, pp. 2 and 10).

The primary purposes for running the test are as follows:

1. Operate the L2E-3 fuel cartridge at peak temperatures of 2500°F and 2600°F for 100 hours or more at each temperature level to evaluate the effectiveness of the ZrO_2 coating against hydrolysis and the release of fission products.

2. Operate the insert fuel cartridge at various temperature levels at specified intervals during the endurance testing to determine fission product release as a function of insert temperature.

3. Obtain additional information pertaining to the effectiveness of an electrostatic precipitator in removing fission products from the reactor effluent (Pincock, 1960d, p. 9).

The peak temperature of the insert was maintained at approximately 2500°F for 101.67 hours and 2600°F for 102.05 hours. The reactor operated 270.97 hours above 100 kW and accumulated 3025.14 MW-h during the test.

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1960d, pp. 18-21) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power level for each run of the test series for the time period during which the reactor operated at a power level greater than 100 kW. The insert was assumed to generate 1.8% of the total reactor power (Pincock, 1960d, p. 15).

Airborne Radioactivity Releases. The radioactivity released during IET #19 was assumed to be primarily composed of fission products, uranium, and Ar-41. Releases were evaluated for the winter season (February 17 to 29) and the growing season (March 1 to April 30). The fission product release was calculated for each of the runs in the test series when the reactor exceeded 100 kW. Uranium releases were estimated on the basis of measured beryllium releases. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Release--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). The airborne emissions from this test were modeled in two parts because testing occurred during both the winter season and growing season.

On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 5.43E-5 Ci/h of I-133 per curie of I-133 inventory in the insert at the end of a test run. This value, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The ratio of the release fraction for noble gases to the

release fraction for radioactive solids was assumed to be 2000:1. This ratio was chosen based upon the reported effectiveness of the zirconia coating in reducing radioactive emissions from the reactor. In an iterative process, the values for the noble gas and radioactive solids release fractions were selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the reported total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product release for the test runs based upon spot sampling data are summarized in Pincock (1960d, p. 134) and are reported as 10-minute-decayed curies. The total fission product release reported for IET #19 was 2892 Ci. This total did not include any release for March 11 operations. The total fission product release for this date was estimated based highest sustained release rate reported for the previous day of operations. The total fission product release for IET #19 was estimated to be 311 Ci during the winter season and 2600 Ci during the growing season.

Uranium Releases--The estimated uranium release during IET #19 was based on measured release rates of beryllium in the airborne effluent (Pincock, 1960d, p. 101). Because the fuel tubes were composed of a mixture of beryllium and uranium, the abrasion of material from the walls of the fuel tubes would release uranium in proportion to the amount of beryllium released. A total of 0.0103 g of 93.4% enriched uranium was estimated to be released during the winter season and 0.0497 g during the growing season.

Argon-41 Releases--A total of 1030 Ci of Ar-41 was estimated to have been released during the winter season and 7440 Ci during the growing season. The Ar-41 production rate was assumed to be 2.8 Ci/MW-h.

IET #20 (May 14 through June 10, 1960)

IET #20 was performed between May 1 and June 13, 1960. Releases of airborne radioactivity were documented to have occurred between May 14 and June 10, 1960, when the reactor was operated at power levels exceeding 100 kW. This test series involved the evaluation of the L2E-2 insert cartridge.

The insert fuel cartridge was composed of hexagonal BeO fueled and unfueled tubes. The fueled tubes were coated on the inner surface with approximately .0015 inch of Al_2O_3 [alumina].

The primary objectives for running the test were as follows:

1. Operate an Al_2O_3 coated BeO insert fuel cartridge at a peak temperature of 2500°F for at least 25 hours and then increase the peak temperature to 2600°F for 100 hours.

2. Evaluate the performance of such an assembly with respect to aerothermodynamics, structural integrity, fission product release, and hydrolysis.

3. Obtain additional information pertaining to the effectiveness of a precipitator in removing fission products from the reactor effluent (Foster et al., 1960, p. 8).

The insert was operated for 45.68 hours at a peak insert temperature of approximately 2500°F and 99.16 hours at approximately 2600°F peak insert temperature (Foster et al., 1960, p. 16).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Foster et al. (1960, pp. 19-20) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a power level greater than 100 kW. The insert was assumed to generate 2.3% of the total reactor power (Foster et al., 1960, p. 35).

Airborne Radioactivity Releases. The radioactivity released during IET #20 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 100 kW. Uranium releases were estimated on the basis of calculated and measured release rates from previous IET tests. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93

through A-97). On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 4.42E-5 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run. This value, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 2000:1. This ratio was chosen based upon the reported effectiveness of the alumina coating on the fuel tubes in reducing radioactive airborne emissions from the reactor. In an iterative process, the values for the noble gas and radioactive solids release fractions were selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product releases for test runs based on spot sampling data are summarized in Foster et al. (1960, p. 78) and are reported as 10-minute-decayed curies. The total fission product release for IET #20 was reported as 5119 Ci.

Uranium Releases--The estimated uranium release during IET #20 was based on uranium release rates observed during past tests. The uranium release rate measured during IET #14 testing for uncoated ceramic fuel was less than 1 μg/s. Based upon the measured release rate of beryllium, the uranium release rate during IET #17 testing of alumina coated ceramic fuel was estimated to be less than 0.8 μg/s. Using an assumed release rate of 1 μg/s, a total of 0.61 g of 93.4% enriched uranium was estimated to be released during IET #20.

Argon-41 Releases--A total of 5130 Ci of Ar-41 was estimated to have been released during IET #20 on the basis of a production rate of 2.8 Ci/MW-h.

IET #21 (FEET #1) (June 29 through August 6, 1960)

IET #21, also referred to as the Fuel Element Effluent Test 1 (FEET #1), was performed between June 20 and August 8, 1960. Releases of airborne

radioactivity were documented to have occurred between June 29 and August 6, 1960, when the reactor was operated at power levels exceeding 100 kW. This test series involved the evaluation of the L2A-2 insert cartridge.

> The insert fuel cartridge consisted of uncoated fueled and unfueled BeO tubes (Pincock, 1960b, p. 8).
>
> The primary purposes for performing this series of tests were as follows:
>
> 1. Provide a source suitable for evaluating the effectiveness of the electrostatic precipitator, which is in the side loop at IET, as a fission product filter. The test was aimed at finding a material which when injected upstream of the precipitator would give 90% removal efficiency.
>
> 2. Obtain further information pertaining to the release of fission products as a function of temperature from uncoated BeO fueled tubes.
>
> 3. Study atmospheric diffusion of fission products released from the IET stack under various meteorological conditions.
>
> Initially it was intended that the insert would only be used for 50 hours at a peak insert temperature of approximately 2700°F. After using up 47.66 hours [during IET #21], the CTF was returned to the Hot Shop and the insert fuel cartridge was removed and examined. The examination showed that the cartridge was still in good condition. Following the LIME experiment, IET 22, it was decided to continue the FEET testing since further information was needed to evaluate the electrostatic precipitator. Testing continued [during IET #23] on the D101-L2A-2 for an additional 45.99 hours (Pincock, 1960b, p. 8).
>
> The majority of the test hours were accumulated with blower operation only. The two aftercooling blowers were operated on high speed to provide the reactor cooling air. This type of operation was accepted as a deviation from the test program and was adopted as a means of eliminating the engine combustion products from the effluent which had definite effects on the precipitator performance. Approximately the same concentration of fission products was maintained in the side loop [to the electrostatic precipitator] under this type of operation as with the engine operation (Pincock, 1960b, p. 18).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1960b, pp. 19-21) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer

code. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a power level greater than 100 kW. The insert was assumed to generate 2.14% of the total reactor power (Pincock, 1960b, p. 38).

Airborne Radioactivity Releases. The radioactivity released during IET #21 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 100 kW. Uranium releases were estimated on the basis of calculated and measured release rates from previous IET tests. Argon-41 was produced in proportion to the energy produced by the reactor.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 1.14E-4 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run. This value, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 1000:1. This ratio was chosen because uncoated fuel tubes were used in the insert. In an iterative process, the values for the noble gas and radioactive solids release fractions were then selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product releases for test runs based on spot sampling data are summarized in Pincock (1960b, p. 96) and are reported as 10-minute-decayed curies. The total fission product release for IET #21 was reported as 2688 Ci. The data contain no explicit estimate for the fission product release on August 2. Based on the highest sustained release rate for reactor operations on the

previous day, the fission product release on August 2 was estimated to be 89 Ci, making the total 2777 Ci.

Uranium Releases--The estimated uranium release during IET #21 was based on uranium release rates observed during past tests. The uranium release rate measured during IET #14 testing for uncoated ceramic fuel was less than 1 μg/s. Based upon the measured release rate of beryllium, the uranium release rate during IET #17 testing of alumina coated ceramic fuel was estimated to be less than 0.8 μg/s. Using an assumed release rate of 1 μg/s, a total of 0.31 g of 93.4% enriched uranium was estimated to be released during IET #21.

Argon-41 Releases--A total of 975 Ci of Ar-41 was estimated to have been released during IET #21 on the basis of a production rate of 2.8 Ci/MW-h.

IET #22 (LIME) (August 25, 1960)

IET #22, also referred to as the Limited Melt Experiment (LIME), was conducted between August 12 and 25, 1960. The reactor was operated on August 22 and 23 in preparation for the LIME experiment performed on August 25. This test series involved the evaluation of the L2E-4 insert cartridge.

> The insert fuel cartridge consisted of fueled and unfueled BeO tubes ... The center tube was an instrumentation tube. Around the center tube were two rows of fueled tubes, eighteen in total, which were blocked to any air flow.
>
> The purpose of the test was to operate the ceramic cartridge for ten minutes at a power level sufficient to cause portions of the plugged fuel region to melt and to evaluate the nature and propagation of such a melt and verify our ability to predict such phenomena.
>
> During the melt portion of the test, the electrostatic precipitator was operated to evaluate its effectiveness in removing fission products from the effluent air and numerous samples were obtained in the exhaust ducting and in the field downwind from IET. These samples were used to determine the fission product release and the diffusion of the released activity

in the atmosphere downwind from IET under existing meteorological conditions (Pincock, 1960c, p. 9).

The reactor operation on August 22 and August 23 resulted in an accumulation of 7.77 megawatt hours, total reactor power. A two day waiting period was required to allow the short lived fission products to decay before attempting to melt the blocked fueled tubes. On August 25, the reactor was taken to approximately 12 MW, total reactor power, on a period of approximately 40 seconds and this power level was held for 10 minutes. The reactor was deliberately scrammed at this time.

During the melt portion of the test, an additional 1.8 megawatt hours were accumulated on the reactor. This made a total of 9.57 MW hours accumulated on the total reactor during IET 22 (Pincock, 1960c, p. 16).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1960c, pp. 17 and 52) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power on August 22 and 23 for the time period during which the reactor operated at a power level greater than approximately 100 kW. The operation on August 25 was based on a detailed representation of the operation (Pincock, 1960c, p. 52). On this date the reactor was assumed to operate for 10 minutes at a power level of 10.8 MW. At the time of the test, the insert was assumed to generate 1.83% of the reactor power. Postoperation analysis indicated that the insert generate 2.5% of the total power (Pincock, 1960c, p. 61). For this analysis the insert was assumed to generate 2.5% of the total reactor power.

Airborne Radioactivity Releases. The radioactivity released during IET #22 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for the LIME experiment on August 25. No fission product releases were assumed for the limited operations on August 22 and 23. Uranium releases were estimated to have the same release fraction as the radioactive fission product solids. Argon-41 was produced in proportion to the energy produced by the reactor and was assumed to have all been released on August 25.

Fission Product Releases--On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest fraction of the total inventory of I-131, I-133, I-134, and I-135 was recorded as 0.036 for I-133 (Pincock, 1960c, p. 69). However, Pincock (1960c, p. 113) also reports that the release fraction for I-134 was 0.0761, when corrected for the inflow of air through the porous fire-brick lining of the stack. Based on this information, the reported release fraction for I-134 was rounded to 8% and then multiplied by a factor of 1.56 to account for plate-out in the sample line, resulting in an estimated release fraction for the RSAC-4 computer code of 0.125 for halogens.

The conservative assumption was made that all of the noble gases were released during the LIME experiment. Several spot samples were collected during the course of the LIME experiment. Based on these samples, a release of 3250 Ci of mixed fission products, expressed as 10-minute-decayed curies (Pincock, 1960c, pp. 83-84). From the information provided in Pincock (1960c), it was not certain that the mixed fission product release included noble gases; the conservative assumption was made that it did not. Using the stated release fractions of 1 for noble gases and 0.125 for halogens, the release fraction for radioactive solids was selected so that the total release of radioactive halogens and solids equaled 3250 Ci, expressed as 10-minute-decayed curies. The total fission product release calculated by the RSAC-4 computer code was 10,500 Ci, expressed as 10-minute-decayed curies.

Uranium Releases--The uranium released during IET #22 was calculated using the same release fraction (0.00452) as assumed for radioactive fission product solids in the RSAC-4 computer code. A total of 0.87 g of 93.4% enriched uranium was estimated to be released.

Argon-41 Releases--A total of 26.8 Ci of Ar-41 was estimated to have been released during IET #22 on the basis of a production rate of 2.8 Ci/MW-h. After accounting for radioactive decay during transport 22.2 Ci of Ar-41 remained at the INEL Site boundary.

IET #23 (FEET #2) (September 7 through October 14, 1960)

IET #23 was performed between September 1 and October 14, 1960. Releases of airborne radioactivity were documented to have occurred between September 7 and October 14, 1960, when the reactor was operated at power levels exceeding 100 kW. This test series involved the continued evaluation of the L2A-2 insert cartridge that had previously been used during IET #21.

> The insert fuel cartridge consisted of uncoated fueled and unfueled BeO tubes.
>
> Following the LIME experiment, IET 22, it was decided to continue the FEET testing since further information was needed to evaluate the electrostatic precipitator. Testing continued [during IET #23] on the D101-L2A-2 for an additional 45.99 hours (Pincock, 1960b, p. 8).
>
> The majority of the test time during IET 23 was accumulated with the two aftercooling blowers providing the coolant air. This was the same procedure used during FEET 1 (Pincock, 1960b, p. 25).
>
> Near the end of the operating period, the insert temperature was increased to a peak of 2800°F [for about 1 hour on October 12, 1960]. This test was performed to obtain field data and also to determine the effect of temperature on the fission product release rate. This was the highest temperature attained during the test (Pincock, 1960b, p. 8).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1960b, pp. 22-24) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code and included the contributions from earlier testing of the insert during IET #21. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a power level greater than 100 kW. The insert was assumed to generate 2.14% of the total reactor power (Pincock, 1960b, p. 38).

Airborne Radioactivity Releases. The radioactivity released during IET #23 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 100 kW. Uranium releases were

estimated on the basis of calculated and measured release rates from previous IET tests. Argon-41 was produced in proportion to the energy produced by the reactor during each period of operations.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). On the basis of the analysis of charcoal trap samples from the 80-ft level of the stack, the largest of the average inventory-normalized release rates for I-131, I-133, and I-135 was determined to be 3.80E-4 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run. This value, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases. The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 1000:1. This ratio was chosen because uncoated fuel tubes were used in the insert. In an iterative process, the values for the noble gas and radioactive solids release fractions were then selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data. The estimated total fission product releases for test runs based on spot sampling data are summarized in Pincock (1960b, p. 97) and are reported as 10-minute-decayed curies. The total fission product release for IET #23 was reported as 2125 Ci. The data contain no explicit estimate for the fission product release on October 14. Based on the highest sustained release rate for reactor operations on the following day, the fission product release on October 14 was estimated to be 40.5 Ci, making the total 2166 Ci.

During IET #23, a 1-hour test was performed on October 12, 1960, in which the insert was operated at peak fuel tube temperature of 2800°F, exceeding the nominal peak temperature of 2700°F applied at other times during the IET #23 operations. Particulate release rates were about a factor of 50 higher at 2800°F than at 2700°F; however, the charcoal trap sampling data did not consistently reflect an increase in the iodine release rates. For this reason, the operation of October 12 was divided into two parts. During the

first 3.17 hours of operation before the temperature was raised to 2800°F, the inventory-normalized release rate was assumed to be the same as applied for all the other test runs (3.80E-4 Ci/h per curie of inventory). For the hour during which the insert was operated at 2800°F, the ratio of the noble gas release fraction and the halogen release fraction was assumed to be 3. This ratio is the average of the values used in the RSAC-4 computer code calculations for the operational releases between September 28 and October 14. This assumption results in a halogen release fraction almost 20 times larger than estimated for the lower temperature operations.

Uranium Releases--The estimated uranium release during IET #23 was based on uranium release rates observed during past tests. The uranium release rate measured during IET #14 testing of uncoated ceramic fuel was less than 1 μg/s. Based upon the measured release rate of beryllium, the uranium release rate during IET #17 testing of alumina coated ceramic fuel was estimated to be less than 0.8 μg/s. Using an assumed release rate of 1 μg/s, a total of 0.31 g of 93.4% enriched uranium was estimated to be released during IET #23.

Argon-41 Releases--A total of 724 Ci of Ar-41 was estimated to have been released during IET #23 on the basis of a production rate of 2.8 Ci/MW-h.

IET #24 (LIME-II) (October 26, 1960)

IET #24, also referred to as the second Limited Melt Experiment (LIME-II) and Sub-LIME, was performed between October 17 and October 26, 1960. The reactor was operated at a power level in excess of 90 kW on October 21 and 24 in preparation for the LIME-II experiment on October 26. This test series involved the evaluation of the L2E-5 insert cartridge.

> The experiment involved a ceramic, BeO, insert fuel cartridge which was designed to simulate a condition in the D140E1 reactor where the air flow through a bundle of 18 tubes was restricted to 10% of the normal flow. The reduced flow region was surrounded by two rings of fueled tubes with normal air flow.

> The test was scheduled to run for two hours or until specified temperature or fission product release limits were reached. Since none of the prescribed limits were attained, the test went the full two hours. During this period of operation, all of the insert temperatures, except for the six fuel temperatures in the reduced flow region which were above the melting point of the thermocouples, showed very steady readings. The fission product release, as recorded by the continuous effluent monitors, peaked at about 1800 curies per hour, age 10 minutes, early in the run and continued to decrease as the test progressed.
>
> The post-operation examination following the experiment showed no evidence of fuel tube melting. There were, however, a large number of tubes fused together in the reduced flow region and in the ring of fuel tubes surrounding this section from stages 6 through 11 (Pincock 1960a, p. 9).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Pincock (1960a) and Baker (1961) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate for the recorded average power on October 21 and 24 for the time period during which the reactor operated at a power level greater than 90 kW (Pincock, 1960a, p. 17). The operation on October 26 was based on a detailed graph of the power operation (Baker, 1961, p. 29). Information presented in Pincock (1960a, p. 41) indicated that the fraction of the reactor power produced by the insert was a minimum of 1.3 to 1.5%. A comparison of the reported insert inventories for several radionuclides (Pincock, 1960a, pp. 49-50) and the RSAC-4 computer code inventory calculations indicated that the insert power was about 2% of the reactor power. A value of 2.2% was assumed in this analysis to ensure conservatism.

Airborne Radioactivity Releases. The radioactivity released during IET #24 was assumed to be composed of fission products, uranium, and Ar-41. The fission product release was calculated for the LIME-II experiment on October 26. No fission product releases were assumed to occur during the limited operations on October 21 and 24. Uranium releases were estimated to have the same release fraction as the radioactive fission product solids. Argon-41 was produced in proportion to the energy produced by the reactor and was assumed to have all been released on August 25.

Fission Product Releases--Because of the apparent inconsistency in the release fractions measured in the charcoal trap samples from the 80-ft level of the stack and the large differences between the values measured in the stack and in the 76-in. duct, the larger and more conservative release fractions measured in the 76-in. duct were used to estimate the halogen release fractions for the LIME-II experiment. The largest release fraction for I-131, I-133 and I-135 was 0.0135 (Baker, 1961, p. 44). The halogen release fraction used in the RSAC-4 computer code was increased to 3% to ensure a conservative estimate.

The integrated release of fission products determined from spot sampling data was reported to be 1880 Ci (Pincock, 1960a, pp. 61-62) and was assumed to be expressed as 10-minute-decayed curies. As in IET #22 (LIME), it was not certain that the reported fission product release included noble gases. The conservative assumption was made that it did not. The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 4000:1. In an iterative process, the values of the release fractions for the noble gas and radioactive solids were selected so that the total release of radioactive solids and halogens, expressed as 10-minute-decayed curies, equaled 1880 Ci. The total fission product release calculate by the RSAC-4 computer code, expressed as 10-minute-decayed curies, was 7650 Ci.

Uranium Releases--The uranium released during IET #24 was calculated using the same release fraction (0.000082) as assumed for radioactive fission product solids in the RSAC-4 computer code. A total of 0.0144 g of 93.4% enriched uranium was estimated to be released.

Argon-41 Releases--A total of 75.9 Ci of Ar-41 was estimated to have been released during IET #24 on the basis of a production rate of 2.8 Ci/MW-h. After accounting for radioactive decay during transport 65.2 Ci of Ar-41 remained at the INEL Site boundary.

IET #25 (November 22 through December 15, 1960)

IET #25 was performed between November 15 and December 16, 1960, and was an extension of the Phase II testing program conducted in IET #18. Releases of airborne radioactivity, corresponding with the significant periods of operation, were assumed to have occurred between November 22 and December 15, 1960.

> The purposes of test series IET No. 25 were to demonstrate the capabilities of the fuel elements above design temperatures and to confirm that the power plant could achieve a full nuclear start as predicted. The reactor went critical on November 14, 1960, and the test program was completed on December 19, 1960.
>
> The first test conducted, an increasing power test, compared system parameters with those on previous tests. After a series of runs to obtain steady-state performance maps of reactor and engine parameters, full nuclear start was made on December 12, 1960. Subsequently, two more nuclear starts were made. The reactor control system, together with a proficient operator, was able to compensate for the transients involved in the nuclear start. The procedure for the start, which was established with the aid of an analog computer, worked very well, and the system behavior closely duplicated that anticipated by the analog studies.
>
> Following the second nuclear start, the reactor was maintained at a power of approximately 29 megawatts for one hour, and then was manually scrammed. One after-cooling blower supplied a flow of 8.6 pounds of cooling air per second to the reactor after scram. Transient recordings were made of selected system parameters. All temperatures started to decline after the scram and, in all cases, the trend established during the first 10 minutes following the scram was maintained for the remainder of the recording period of one hour.
>
> A total of 20.3 hours of elevated performance was accumulated at the conclusion of the full nuclear starts. Elevated performance was defined as full nuclear operation with a peak fuel element temperature of 2030°F $\pm$ 20°F. All reactor variables remained quite stable during this run and the reactor operation was completely normal. At the termination of this operation, the instrumentation indicated that the reactor was fully capable of continued operation (Linn et al., 1962, pp. 41-42).

Fission Product Inventory. The inventory of radioactive fission products in the reactor was reconstructed using the available reactor operating history recorded in Highberg et al. (1961, pp. 14-19). The history

was incomplete, lacking the time of day that operations started and ended and the length of reactor operating times. The IET #18 operational history (Highberg, 1960, pp. 7-12) was reviewed to determine the similarities with the IET #25 operations; the two operations were quite similar. During the 16-day period of IET #18 testing from January 23 to February 7, 1960, there were 14 days of operations at a power level above 0.01 MW accomplished in 275 runs. During the 32-day period of IET #25 testing from November 15 to December 16, 1960, there were 17 days of power operations above 0.01 MW accomplished in 242 runs. From data given for the IET #18 test series, the length of a run ranged from approximately 40 to 60 minutes. For reconstructing the operational history of IET #25, all runs were conservatively assumed to have lasted for 60 minutes.

Airborne Radioactivity Releases. The radioactivity released during IET #25 was assumed to be composed of fission products, uranium, and Ar-41. Consistent with IETs #16 and #18, the released fission products were assumed to originate from uranium remaining in the cocoon of the reactor as a result of the nuclear excursion that occurred at the end of IET #13. Releases were estimated for the power operations documented in Highberg et al. (1961, pp. 14-19).

Fission Product Releases--Only the following brief summary of effluent monitoring activities and results was available:

> Continuous effluent monitoring was maintained to measure and record the activity released to the atmosphere by the power plant. The maximum output was 3.4 curies/hour (measured 10 minutes after release). The total output for the test series was 218 curies (measured 10 minutes after release). The maximum release rate for I-131 was approximately 0.7 curies/hour (measured 10 minutes after release). The total offsite inhaled and ingested dose was below measurable amounts during this test series (Highberg et al., 1961, p. 153).

As described in the IET #18 discussion, the reported release rate of I-131 is not consistent with the reported release rate for total fission products nor is the statement consistent with the iodine sampling results of IET #16. In the event 0.7 Ci/h for I-131 was a correct value, then the total

iodine release should have been in the range of 7 to 70 Ci/h. There were no milk samples taken during, or within 10 days after, the period of IET #25, which is also inconsistent with an I-131 release rate of this magnitude. Because the effluent monitoring paragraphs for the reports for IETs #18 and #25 are worded exactly the same, except for the numerical values, it is possible that an error with reference to I-131 made in the IET #18 report was carried over into the IET #25 report. Comparing IETs #18 and #25, the lower effluent release rates for this test are consistent with a lower amount of uranium that might be expected to be in the cocoon because of ablation during the IET #18 operation and/or because of decontamination following IET #18.

In summary, because the erroneous reference to I-131 is believed to have been a carryover from IET #18 and that either (a) the release rate of I-131 should have been expressed as mCi/h rather than Ci/h, or (b) the reference to I-131 should have been to I-135, the more conservative of the two possibilities was chosen. The assumption was made that I-135 should have been the isotope of iodine to which reference was made rather than I-131. The ratio of total fission product activity release rate to I-131 release rate was reported to be the same for IETs #18 and #25. This is reasonable because the release was assumed to originate with the uranium in the cocoon of the reactor. In reconstructing the IET #25 source term, an attempt was made to use the same methodology as used for IET #18. This technique produced unreasonable results; the halogen release fractions were larger than the noble gas release fractions. Therefore, based on engineering judgment, the release fractions for use in the RSAC-4 computer code for IET #25 were chosen as described below. The iodine release fraction was calculated to preserve the documented 0.7 Ci/h iodine release rate but for I-135 rather than I-131, the release fraction for noble gases was set equal to that for the halogens, and the solids release fraction was set at 1/2000 the respective halogen release fraction. The release fraction associated with the 0.7 Ci/h release rate for I-135 was assumed to apply to the period of operation and was reduced for other periods of lower power operation in proportion to the energy produced for that respective period.

Uranium Release--Uranium releases were not addressed in the operations report for IET #25. To estimate the uranium released, the IET #16

uranium release was scaled upwards in proportion to the nuclear energy generated during the test series. Thus, the uranium release was estimated to be 0.24 g of 93.4% enriched uranium.

Argon Release--A total of 10,000 Ci of Ar-41 was estimated to have been released during IET #25 on the basis of a production rate of 2.8 Ci/MW-h.

IET #26 (December 23, 1960 through March 30, 1961)

IET #26 was performed between December 22, 1960, and March 31, 1961. Releases of airborne radioactivity were assumed to occur between December 23, 1960, and March 30, 1961, when the reactor was operated at power levels exceeding 120 kW. This test series involved the evaluation of the L2E-6 insert cartridge.

> The insert cartridge consisted of fueled and unfueled BeO hex tubes. The fueled tubes were coated on the inner surface with ZrO_2.
>
> The primary purposes for performing this test series were as follows:
>
> 1. Evaluate the ZrO_2 coating at temperatures above ACT [presumably referring to the Advanced Core Test] design conditions. The duration of the test was expected to be between 100 and 500 hours. The test did in fact extend over some 350 hours of high temperature operation,
>
> 2. Operate the insert at a fuel temperature ranging from 2500 to 2800°F in order to better understand the fission product release behavior as a function of time and temperature (Field, 1961, p. 6).
>
> A total of 352.39 hours of testing was obtained at or above a peak insert fuel element temperature of 2520°F. Of this time, 3.08 hours were accumulated at 2750°F, 196.71 hours at 2700°F, 102.08 hours at 2600°F and 50.52 hours at 2520°F (Field, 1961, p. 12).

During the examination of the insert following the test, a piece of foreign material was lodged on the upper grid plate of the insert, and it probably restricted air flow within part of the grid plate. The impact of this on airborne effluent releases is discussed in Field (1961, p. 48).

> In light of the post-operational examination, which revealed a portion of the tubes partially blocked with a piece of fibre tape (thought to have been deposited when the cold riser on the CTF failed) the release data appears to show about a factor of 2 to 3 increase at this time (~123 test hours). This is not clear cut but [from] examining the data at any of the test conditions [this] appears to be the case.

From a review of the insert temperature profiles, it was likely that the tube blockage occurred on or about January 30 (Field, 1961, p. 31).

Fission Product Inventory. The inventory of radioactive fission products in the insert was based on the reactor operating history recorded in Field (1961, p. 13-17) and the fraction of the reactor power produced by the insert. Inventory calculations were performed with the RSAC-4 computer code. The reactor was assumed to operate at the recorded average power for each run of the test series for the time period during which the reactor operated at a power level greater than 120 kW. The insert was estimated to produce 200 kW (Field, 1961, p. 27) or 2.5% of the total reactor power at endurance run conditions of about 8.0 MW.

Airborne Radioactivity Releases. The radioactivity released during IET #26 was assumed to be composed of fission products, uranium, and Ar-41. Releases were evaluated for the winter season (December 23, 1960 to February 28, 1961) and the growing season (March 1 to 30, 1960). The fission product release was calculated for each of the runs in the test series when the reactor power exceeded 120 kW. Uranium releases were estimated on the basis of calculated and measured release rates from previous IET tests. Argon-41 was produced in proportion to the energy produced by the reactor during each period of operations.

Fission Product Releases--The halogen release fractions used in the RSAC-4 computer code for estimating the releases of radioiodines were calculated as described in the discussion of charcoal trap samples (pp. A-93 through A-97). The monitoring data for the charcoal trap samples from the 80-ft level of the stack were carefully reviewed because of the piece of debris that was blocking the air flow to some tubes in the insert. The blockage, which occurred on or about January 30, did affect the release rates;

however, a trend toward higher inventory-normalized release rates was also was noted on or about March 1. On the basis of this analysis, the airborne releases of radioactive isotopes of iodine were considered in two periods, the winter season (December 23, 1960 through February 28, 1961) and the growing season (March 1 through 30). The largest of the average inventory-normalized release rates for I-131, I-133, and I-135 during the winter season was determined to be 2.05E-5 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run. The largest of the average inventory-normalized release rates for I-131, I-133, and I-135 during the growing season was determined to be 5.26E-5 Ci/h of I-133 released per curie of I-133 inventory in the insert at the end of a test run. These values, when multiplied by the individual test run operating times, represented the halogen release fractions used in the RSAC-4 computer code to estimate iodine releases.

The ratio of the release fraction for noble gases to the release fraction for radioactive solids used in the RSAC-4 computer code was assumed to be 2000:1. This ratio was chosen based upon the reported effectiveness of the zirconium oxide coating on the fuel tubes in reducing radioactive airborne emissions from the reactor. In an iterative process, the values for the noble gas and radioactive solids release fractions were then selected so that the total fission product release calculated using the RSAC-4 computer code equaled or slightly exceeded the total fission product release estimated for individual runs from the spot sampling data.

A comprehensive listing of the total fission product releases for test runs based on spot sampling data are not provided. The results of monitoring for fission products are summarized in Field (1961, p. 61).

> Observed effluent release rates for this test were in general lower than those observed in any previous ceramic insert test performed at IET. During the course of the test while operating at an insert fuel element temperature of 2720°F the release rate, indicated by spot samples taken in the vault, increased from approximately 4 curies/hour to 15-18 curies per hour at the completion of the test. Just prior to the test completion the release rates at the various "on test" conditions were observed as follows: at 2520°F approximately 4.4 curies/hour, 2620°F

approximately 7.0 curies/hour and 2720°F from 15 to 18 curies/hour.

The release rate for March 9 was reported as 10.0 Ci/h and average release rates for March 7, 8, and 17 were calculated to be 8.0, 13.3, and 16.3 Ci/h, respectively (Field, 1961, p. 67). Using this information, total fission product releases were estimated for each of the runs in the test series. A total of 1510 Ci was estimated to have been released during the winter season and 1500 Ci during the growing season. In accordance with standard practice in previous IETs, the releases were assumed to be expressed in terms of 10-minute-decayed curies.

During IET #26, charcoal trap samples were collected to assess releases of radioactivity during reactor startup and shutdown (Field, 1961, pp. 49-58). These releases were postulated to be the result of diffusion or erosion of radioactivity from fuel elements and the erosion of plated out fission products from surfaces in the effluent ducting (Field, 1961, p. 49). An analysis of this data indicated that the overall source term for radioactive isotopes of iodine would be increased by at most 5% during IET #26 if these releases were considered. This additional amount of radioactivity was small compared to the uncertainties inherent in the effluent monitoring data and the conservative assumptions used to estimate the operational releases. Therefore, further explicit treatment of the radioactive emissions during reactor startup and shutdown was not warranted.

Uranium Releases--The estimated uranium release during IET #26 was based on uranium release rates observed during past tests. The uranium release rate measured during IET #14 testing of uncoated ceramic fuel was less than 1 μg/s. Based upon the measured release rate of beryllium, the uranium release rate during IET #17 testing of alumina coated ceramic fuel was estimated to be less than 0.8 μg/s. Using an assumed release rate of 1 μg/s during IET #26, a total of 1.03 g of 93.4% enriched uranium was estimated to be released during the winter season. A total of 0.406 g of 93.4% enriched uranium was estimated to be released during the growing season.

Argon-41 Releases--A total of 6510 Ci of Ar-41 was estimated to have been released during the winter season and 2590 Ci during the growing season of IET #26. These values were calculated on the basis of a production rate of 2.8 Ci/MW-h.

OMRE Solvent Burning Experiment (November 16, 1960)

> An experiment to determine the feasibility of open-air burning of contaminated solvents accumulated at the Organic Moderated Reactor Experiment [OMRE] facility was conducted on November 16, 1960.
>
> Approximately 400 gallons of liquid were placed in an open vessel and ignited. Lapse conditions accompanied by a strong steady wind (25 mph) prevailed at the time of the test. The liquid was composed of diesel oil, xylene, methychlor (Methylchloroform), and a small amount of water.
>
> The contaminants and their percentages were determined to be Mn-54 (60%), Co-60 (30%), and Fe-59 (10%) (AEC, 1961).

The radioactivity concentration in the xylene, the highest concentration of the three liquids, was 1.7E-8 Ci/mL. The quantity of radioactivity released was less than 0.026 Ci. The OMRE Solvent Burning Experiment was evaluated by comparison to other dose calculations. The EDE was insignificant compared to the annual releases for 1960.

1959 ICPP Criticality (October 16, 1959)

At approximately 3:00 a.m. on October 16, 1959, a criticality occurred at the ICPP in vessel WH-100 (Ginkel et al., 1960). The magnitude of the criticality was estimated to be less than 4E+19 fissions. Releases to the environment consisted primarily of iodines, noble gases, and the short-lived daughters of the noble gases.

> Available evidence indicates 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 the most likely mechanism by which the transfer took place (Ginkel et al., 1960, p. 1).

At the time of the incident, the meteorological conditions that prevailed appear to have been a strong lapse condition with a wind out of the north with a wind speed of 20 to 30 mi/h (Ginkel et al., 1960, p. 40).

Iodine releases were based on the measured release of 3.73 Ci of I-131. The quantity of I-132 initially reported to have been released to the environment was found to be in error. Original calculations failed to account for the precursor, Te-132. Current calculations the RSAC-4 computer code show that it would have been impossible to have produced the quantity of I-132 originally reported to have been released from the criticality. Thus, releases of I-132 and other iodines were assumed to be at the same fraction of the total produced in the criticality as measured for I-131.

All of the noble gases produced during the criticality were assumed to be released. No measurable quantities of particulate activity were released. However, the particulate daughters of the short-lived noble gases were assumed to grow into equilibrium with their parent after release.

The radioactivity released to the environment from this criticality has been calculated using the RSAC-4 computer code. Activities reported have been decayed 0.52 hour, the time calculated for the plume to reach the INEL Site boundary.

ICPP Plutonium Release (July 9 through 11, 1959)

Approximately 105 mCi of plutonium were released to the atmosphere at ICPP. The release was the result of burning plutonium contaminated waste solvent. Monitoring of this incident is described in AEC (1960, pp. 127-128). The release was not modeled.

Collection Tank Release (December 1958)

Approximately 1 Ci of radioactive noble gases and iodine was released to the atmosphere from a liquid waste tank at the ICPP because of a leaking flange (ERDA, 1977). The release is included in the operational releases.

Solvent Burner Release (July through September 1958)

Approximately 0.25 Ci of long-lived particulate activity was released from the ICPP solvent burner via the main stack (ERDA, 1977). The release was included in the operational releases.

FECF Filter Break (October 29 through 30, 1958)

Fuel elements sent to the ICPP contained structural materials on the ends of the elements. This structural material, commonly called end pieces, was cut off from the section of the element containing the uranium before being sent to the CPP-601 facility for processing. This cutting operation was done in the Fuel Element Cutting Facility (FECF) located in the CPP-603 facility. During the night of October 29, 1958, and early on the morning of October 30, 1958, decontamination operations were conducted in the FECF. Acid fumes from the decontamination operations caused failure of the exhaust filters in the FECF resulting in the release of particulate activity to the south of the ICPP.

Approximately 100 Ci of long half-life particulate activity was released over an area of approximately 200 acres (AEC, 1959). Measurements of the ground level contamination were taken following the incident using a Geiger-Mueller survey instrument at a distance of 3 ft above the ground using an "open window" on the instrument. The open window made the instrument responsive to beta as well as gamma radiation. Thus, the instrument reading would relate to the dose rate to the skin. At a distance of 0.6 mi (approximately 1E+3 m) the open window instrument reading was 0.07 mR/h (AEC, 1960).

The majority of the activity released from the FECF was associated with very large particles and was rapidly deposited on the ground near the FECF. Some of the released material would also have been small and capable of being transported downwind. Because there were no downwind measurements made of the airborne material, it was necessary to reconstruct the best estimate of the amount of material that remained airborne. An assumption was made that the material that reached the furthermost downwind points on the isopleth

presented in AEC (1960) was small in particle size and was capable of being transported further downwind.

Fuels being cut in the FECF were approximately 1 year old. An RSAC-4 calculation was made to estimate the skin dose from 1 Ci of 1 year old MTR fission products deposited on the ground at 1000 m under the light wind conditions at the time of the accident. This calculation was compared to the 0.07 mrem measured dose rate. The amount of each radionuclide released was raised proportionally so that the calculated skin dose rate equalled the measured dose rate. The major radionuclides predicted by this method match those measured on sagebrush at the time of the accident (AEC, 1959). The RSAC-4 computer code calculation includes other radionuclides that were not detected in the analysis of sagebrush samples.

FPFRTs (July to September 1958)

The Fission Products Field Release Tests (FPFRTs) were conducted at Test GRID III at the INEL. These tests were sponsored by the Air Research and Development Command of the U.S. Air Force from July 25 through September 26, 1958. The tests were performed to obtain information for evaluating the release of radioactivity from potential accidents involving nuclear powered aircraft using metallic fuel. Nine tests, designated FPFRT-1 through -9, were performed to measure fission product release fractions, diffusion of radioactivity in air, and deposition of radioactivity on vegetation and the ground. The radioactivity released in FPFRT-1, -2, -3, -4, and -9 came from aged fuel elements that had been removed from a reactor more than 920 days before the tests. The radioactivity released from the other tests came from fresh fuel elements that had been removed from the reactor 40 to 65 days before the tests (Wehman, 1959; Convair, 1959).

The Idaho Operations Office of the AEC was responsible for ensuring that onsite and offsite radiation doses did not exceed permissible levels during the tests. Because of the potential for affecting adjacent INEL facilities and releasing radioactivity to an offsite location, all tests were performed under strict operational controls based on meteorology and with extensive field monitoring to verify predicted test behavior (Wehman, 1959, p. 2).

The methods used to release the radioactivity are described in Convair (1959, pp. 31-32):

> Melting of the test samples was accomplished with a specially designed induction furnace . . . Induction heating was chosen so that fast melting and adequate temperature control could be obtained. The furnace was equipped with sequentially operated effluent samplers, and metered air was introduced to displace the effluent from the furnace crucible. The furnace, power supply, and operating controls were located at or near the apex of the network.
>
> The criteria established for the release operation were as follows:
>
> - Stage to be melted in air
> - Melt to occur within a few minutes after initial application of power
> - Sustained melt temperature to exist for 10 minutes following initial melting
> - Furnace to be purged with metered air and quantitative samples of the effluent taken throughout the heating cycle
> - Effluent to be released to the environment at five feet above ground level.

The operating histories for each fuel element were recorded in Convair (1959, Appendix N) and were used to estimate the inventory of radioactivity in the elements (Convair, 1959, Appendix A). The fission product inventory in each fuel element was recalculated using the RSAC-4 computer code and used in calculations to determine the amount of radioactivity released during the individual FPFRTs. The inventory calculations made with the RSAC-4 computer code agreed well with those reported in Convair (1959). The dates on which each of the tests were performed and the accumulated decay time since the fuel was removed from the reactor are shown in Table A-30.

Release fractions used to calculate source terms for the FPFRTs were chosen based on the values reported for the FPFRTs and are summarized in Table A-31 (Convair, 1959, Appendix N). The largest release fractions were reported for FPFRT-5. During this test, the induction furnace operated as

Table A-30. Dates of FPFRTs performed at the INEL Site during 1958

Test Number	Date	Time	Time Since Removal From Reactor (d)
FPFRT-1	July 25	18:09	922
FPFRT-2	August 4	20:16	932
FPFRT-3	August 6	19:14	934
FPFRT-4	August 14	18:16	942
FPFRT-5	August 27	17:19	42.9
FPFRT-6	September 4	17:12	50.9
FPFRT-7	September 17	19:04	64
FPFRT-8	September 18	18:32	65
FPFRT-9	September 26	16:22	985

Table A-31. Reported release fractions for FPFRTs[a]

	Release Fraction (%)								
Nuclide	FPFRT-1	FPFRT-2	FPFRT-3	FPFRT-4	FPFRT-5	FPFRT-6	FPFRT-7	FPFRT-8	FPFRT-9
Ba-140					43.				
Ce-141					14.		0.05	0.3	
Cs-137	43.	43.	47.	83.	87.	36.	34.	24.	50.
I-131					100.	50.	50.	50.	
Ru-103					9.	0.75	0.8	0.75	
Sr-89					41.				
Sr-90	1.7		0.03	0.68					0.53
Zr-95					65.		0.25	0.05	

a. Source: Convair (1959).

planned, raising the temperature of the fuel element to the melting point and maintaining that temperature for 10 minutes.

The release fractions reported for FPFRT-5 were conservatively applied to all tests except FPFRT-4 and -8 (see Table A-32). During FPFRT-4 and -8, a thermocouple malfunctioned, resulting in higher furnace temperatures than observed in FPFRT-5. Because of the higher temperatures, the release fractions assumed for these tests were larger than those for FPFRT-5 and are based largely upon professional judgment (see Table A-32). In the case of parent and daughter decay chains, the release fraction for the parent was also assumed to apply to the daughter. Release fractions were not measured and reported for all radionuclides present in the fuel used in the FPFRT tests. In these cases, the release fractions were chosen based on information reported in ANS (1984, pp. 4.3-4.13) or were assumed to be equal to the release fraction measured for Ru (see Table A-32). Based on professional judgment, the release fractions for these radionuclides were increased for FPFRT-4 and -8 because of the more severe test conditions that existed.

Table A-32. Release fractions used for FPFRT source term calculations

	Release Fractions		
Element	FPFRT-4	FPFRT-8	All Others
Ba	N/A	0.47	0.43[a]
Ce	0.17	0.19	0.14[a]
Cs	0.90	0.92	0.87[a]
I	N/A	1.00	1.00[a]
Ru	0.11	0.13	0.09[a]
Sr	0.45	0.47	0.41[a]
Zr	0.67	0.69	0.65[a]
Te	N/A	0.92	0.87
Y	0.45	0.47	0.41
Mo	N/A	0.13	0.09
Pr	N/A	0.13	0.09

a. Release fractions for the indicated element were based on the release fractions reported for FPFRT-5.

Blower Failure (August 1958)

An atmospheric releasee of approximately 10 μCi of long-lived fission product activity occurred at the ICPP because of a blower failure (ERDA, 1977). The release was included in the operational releases.

BORAX-IV Test (March 11 through 27, 1958)

The BORAX-IV reactor was intentionally operated at a power of 2.4 MW with a large number of defective fuel elements. During the tests, radioactive fission products leaked from the fuel causing high radiation and building contamination. Environmental releases were primarily limited to cleanup operations (ERDA, 1977; AEC, 1959). The release was not modeled. Sufficient information was not available to calculate the quantity of radioactive materials released. An examination of the description of the events and the reactor power levels leads the authors to the conclusion that the evaluation of the event would not substantially change the annual doses calculated.

Iodine Release (March 1958)

Approximately 1 Ci of I-131 was released to the atmosphere during a routine waste transfer operation at the ICPP (ERDA, 1977). The release was included in the operational releases.

Fuel Element Burn Test, A and B (March 20, 1957)

The Fuel Element Burn Tests A and B were conducted at GRID III in 1957. Aged fuel assemblies were heated to drive off fission products for the purpose of determining the fission product release in the event of a nuclear aircraft crash involving a fire. The tests, in support of the GE ANP Program, consisted of two experiments designated as Phase A and Phase B (Brodsky and Beard, 1960).

The Phase A experiment, conducted on March 20, 1957, commenced at 2:19 p.m. and involved a stainless steel clad fuel element that had been operated in a reactor generating a total power of 20 MW for a period of 120

hours and then subsequently decayed for a period of 70 days. The note to Table 1, entitled "Fission Product Composition, Phase A," on p. 35 of Brodsky and Beard (1960), states,

> Based on 20 MW elements at 120 hr operation plus 70 days immediate decay. Assuming a gamma energy of 0.7 Mev and a gamma 'curie' content of 2500, the radiation field would be 12,000 r/hr at 1 ft from a point source (Brodsky and Beard, 1960).

The experimental setup for Phase A is described in the same report:

> A 90 per cent enriched fuel element clad in stainless steel was supported in an airframe and surrounded by other reactor materials to simulate conditions of a nuclear airplane crash. The fuel element contained about 5000 curies of fission products. A pool of jet fuel underneath the airframe was ignited and the fire burned for about 2 hr. After the fire died, no activity was measurable by portable survey equipment except inside a 24-sq ft area around Ground Zero. Levels averaged 200 mr/hr at 3 ft above ground within this area. Low-activity levels were detected with air samplers and fall-out plates downwind to 5000 ft, indicating that only an extremely small fraction of the fission products had been released in this experiment. The fuel element was still intact after the fire except for one small puncture, although it had reached a temperature of about 2250°F (Brodsky and Beard, 1960, p. 32).

Meteorological conditions during the Phase A testing were described as

> The 20-ft altitude winds ranged from 200 to 210° at 13 to 14 miles per hour. Thus the cloud moved 20 to 35° to the northwest of the sampling sector most of the time, but short-lived direction changes did bring the cloud directly over the sampling sector at times.
>
> The vertical temperature stratification during the burning period showed temperature decreases from 3 to 4°F from ground to 250 ft, indicating optimum diffusion conditions for this phase (Brodsky and Beard (1960, p. 35)

Phase B involved a fuel element with a curie content estimated at twice that for the Phase A experiment. The operational history of this fuel element was described as 6.16E+21 fissions, with a subsequent decay of 250 days (Brodsky and Beard, 1960, p. 37). During the Phase B experiment, which

commenced at 6:47 p.m. on March 20, 1957, this fuel element was subjected to much higher temperatures than occurred in the Phase A test.

> A fuel element was heated to about 5000°F by an oxygen-fed fire of thermite, steel wool, and iron filings. The fuel element containing 10,000 curies of fission products was supported upright is a cylindrical furnace and surrounded by the combustible materials. Oxygen was fed into the bottom of the furnace through a side tube near the lower part of the element. An arrangement of feeders around the fuel element dispersed the oxygen throughout the furnace at a steady rate during the burning period. The burning continued for about four minutes. Most of the fuel element was melted and dispersed within 90 seconds. Some of the molten uranium and fission products spurted out the top of the furnace in Roman candle fashion, but most of the molten element ran out the side tube to the ground (Brodsky and Beard, 1960, p. 32).

The meteorological conditions during the Phase B testing were described:

> The 20-ft winds at the test site for the first two 15-min periods after 1846 MST averaged 7 miles per hour from 215° and 6 miles per hour from 210°, with a total swing from 195 to 235°. This shows that just downwind of the test site, the cloud moved 20 to 25 degrees north-west of the center line of the sampling sector. The vertical temperature variation had changed to an inversion condition, with temperature increases from 1 to 2 degrees Fahrenheit from ground to 250-ft level. This inversion prevented the cloud from rising to higher levels and gave poor diffusion conditions (Brodsky and Beard, 1960, p. 38).

The reported information was not sufficient to enable direct modeling with the RSAC-4 computer code to calculate the fission product inventory in the fuel elements without making further assumptions. The report (Brodsky and Beard, 1960) states that the fuel element for the Phase A test was operated in a reactor generating a total power of 20 MW. The report provides the calculated fission product inventory and also states that the inventory was generated during a 120-hour operation with a subsequent decay of 70 days. Therefore, a trial calculation was made with the RSAC-4 computer code assuming that the fuel element generated 1.0 MW for a period of 120 hours with a subsequent decay of 70 days. The RSAC-4-calculated inventory was compared with the inventory provided for this element in Table 1, page 35, of Brodsky and Beard (1960). In an iterative process, the power level was adjusted until

the Cs-137 and Ce-144 inventories were within 10% of those provided in the referenced Table 1. Making this adjustment required the initially assumed operating power level of the fuel element be lowered from 1.0 to 0.568 MW. The resulting RSAC-4 computer code -calculated inventory is compared with the referenced inventory in Table A-33. The assumed fission product release fractions for the Phase A test were conservatively set at 50% for the noble gases, 2.5% for the halogens, and 0.1% for the solids.

For Phase B, again, the data contained in the referenced reports were insufficient for computer modeling with the RSAC-4 computer code without some additional assumptions being made. The report (Brodsky and Beard, 1960)

Table A-33. Referenced and RSAC-4-calculated fission product inventories for the Fuel Element Burn Test Phase A element

Radionuclide	Referenced Inventory[a] (Ci)	RSAC-4-Calculated (Ci)
Nb-95	900	805
Zr-95	600	763
Sr-89	550	584
Sr-90 (Y-90)	10	9.25 (9.26)[b]
Y-91	550	711
Ce-141	500	646
Ce-144 (Pr-144)	250	265 (265)[b]
Ru-103	300	371
Rh-103m	300	370
Ru-106	25	16.4
Cs-137	10	9.38
Ba-140 (La-140)	300	161 (185)[b]
Pr-143	150	199
Nd-147	30	35.7
I-131	25	11.9
Xe-133	20	1.81
Te-129	30	8.84
Miscellaneous plus error	450	105
TOTAL	5000	5530

a. Values from Table 1, Brodsky and Beard (1960, p. 35).

b. Value in parentheses is the activity of the daughter radionuclide

states that the inventory for the Phase B fuel element, given in Table 3, page 37, is based on 6.16E+21 fissions followed by a 250-day decay period. Although the title to Table 3 is "Fission Product Composition, Phase B," the values given in the table are not curie values, but relative yields; and do not add to 10,000 but rather to 105.96. Using the reported fissions (6.16E+21) for the fuel element and applying the conversion factor of 3.12E+10 fission/W-s, the fuel element generated 1.974E+11 W-s of energy, or 1.974E+5 MW-s, while in the reactor. Because the fuel element was also clad with stainless steel, it was assumed that this fuel element was the same type as used in the Phase A test and that it operated in the same reactor as the Phase A fuel element. A reactor operation of 3.475E+5 seconds (4.022 days) at a power level of 0.568 MW would be required to produce the 1.974E+5 MW-s burnup. This derived operation would yield a smaller inventory than the Phase A fuel element, in direct contradiction with information contained in the report. With a 250-day decay applicable to the Phase B fuel element, the inventory would be considerably less than the Phase A inventory, not twice the inventory as reported. Further calculations showed that an operating time of 69 days would have been required at 0.568 MW to build a fission product inventory that would be about twice the Phase A fuel element inventory (or about 10,000 Ci) after a 250-day decay period. This operation was assumed for the calculation of the inventory for the Phase B fuel element and resulted in a calculated fission product inventory of 10,700 Ci. For the reactors that were operational during this time period, the chosen parameters for both the Phase A and Phase B fuel elements are all within reasonable bounds. The fission product release fractions chosen for the Phase B test, because of the severity of the melting and the 5000°F temperature attained by the furnace, were 100% for the noble gases, 50% for the halogens, and 1% for the solids.

The only information provided concerning the uranium in the fuel elements was that it was 90% enriched. To estimate the release of uranium from these tests, each of the fuel elements were assumed to contain 200 g of U-235, which corresponds to a total uranium content of 222 g. The 200 g value represents the nominal U-235 content of a fuel element in the MTR reactor (Brodsky and Beard, 1960, p. 6) and provides a reasonable estimate of the uranium content in a generic fuel element. It was assumed that the release fractions chosen for the solid fission products, 0.1% for Phase A and 1% for

Phase B, also applied to the uranium in the fuel elements. The estimated releases of uranium isotopes during the Phase A test were 1.34E-5 Ci of U-234, 4.32E-7 Ci of U-235 and 6.75E-9 Ci of U-238. The estimated releases of uranium isotopes during the Phase B test were 1.34E-4 Ci of U-234, 4.32E-6 Ci of U-235 and 6.75E-8 Ci of U-238.

EBR-I Core Meltdown (November 29, 1955)

Early in 1954, core II was placed in the reactor and operated intermittently to 1150 kW, the maximum authorized power level. Core meltdown on November 29, 1955, involved the melting of 40 to 50% of the core. Initially, the core contained about 52 kg of U-235 in an enrichment of 93.21% and 10.5 g plutonium foil that is believed, according to metallurgical examinations performed at the Lemont Illinois hot cell facility, to have remained with the core material that went to that facility for analysis (Zinn, 1956; Brittan, 1958; Stratton, 1967).

> How, then, was it known that melting of the fuel had occurred? There was, of course, no unusual noise, explosion, or fire. It is characteristic of this work that the first indication came through radioactivity. After a period of 15 minutes, detectors for radioactivity installed both in the cooling system, some distance from the reactor, and in the ventilation exhaust ducts of the building began to show readings higher than normal. For the detector installed in the cooling system, it could be inferred that gaseous radioactive fission products had diffused through the gas blanket connections to the point where the detector was installed. (It should be recalled that the coolant was not flowing.) The activity in the building exhaust, while low, also was abnormal. This was due to the fact that the thermocouples for this experiment had been installed with temporary connections that did not seal gastight, so that some of the radioactive gas in the reactor vessel diffused into the room. As a precaution, the room was evacuated promptly. Normal use of the room was resumed just as soon as quantitative measurements of the activity could be made. On later examination, it was found that while some of the fuel rods could be removed, many could not be pulled through the holding plate (Zinn, 1956, p. 103).

Because of the low power level of the reactor, the low concentrations detected in the building at the time of the core meltdown, and the generally low releases from this type of reactor, the release was not modeled.

NRF S1W Engineering Test (June 1955)

An engineering test was performed in June 1955 to obtain information on the effects of boiling conditions in naval reactors. This test was conducted using a specially designed and instrumented removable test assembly that was designed to explore fuel element integrity beyond operating limits in effect at the time. This test was conducted in accordance with preplanned procedures and under carefully controlled plant conditions.

During testing on June 18, 1955, 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. The test data were valuable in determining the limits of fuel element performance, which was important in the development of subsequent naval reactor designs.

The amount of radiation exposure to a member of the general public resulting from this test was too small to be measured. The radioactivity released was calculated based on radionuclide inventory in the assembly, measured radiation levels in the primary cooling system, recorded leak rates, and conservative radionuclide release fractions. No credit was taken for removal of radionuclides by the filtering systems installed in the primary cooling system or the radioactivity removed with the water during drain and refill operations conducted during the period, even though these effects are believed to have substantially reduced the actual emissions.

BORAX-I Excursion (July 22, 1954)

The BORAX-I was constructed as a water-cooled, water-moderated reactor during the summer and fall of 1953. The reactor operated with 30 fuel elements of the MTR type, each containing about 140 g of U-235 that was 90% enriched. Nondestructive experiments were conducted in the later portion of 1953 and early summer of 1954. With the completion of these experiments, a final destructive excursion was planned in which about 4% of the fuel plates were anticipated to melt.

After a day of waiting for favorable wind conditions, the test was conducted on the morning of July 22, 1957. The test was described in Dietrich (1954, p. 25).

> In a very short time after the release of the central control rod a column of what appeared to be dark grey smoke was ejected from the reactor to a height of some 80 feet. The difference in appearance between this material and the silvery-white mixture of steam and water which had been ejected during the slower-period 1953 experiments was striking.

Ejected material probably contained burning aluminum as well as the aluminum-uranium fuel alloy. Material was ejected 200 to 300 ft from the reactor, and only about one-third of the core remained within the pit after the excursion. During the excursion, which was estimated to have produced 135 MW-s of energy, major melting of the core occurred in which essentially all of the fuel assemblies were involved. At the time of the destructive excursion, the wind was blowing from the northeast with a wind speed of 8 mph at ground level and 20 mph at 250 ft.

> The air-dispersed material from the reactor was blown in a direction about 35 degrees west of south. Mobile monitoring teams crossing the trajectory of the material at 8:35 AM. (15 minutes after the excursion) at a distance of about 0.8 mi. from the reactor, recorded a maximum reading of 5 mr/hr with open-window survey meter, 3 feet above the ground. A similar reading at 8:45 AM, about 2.3 miles from the reactor, gave 2.0 mr/hr (Dietrich, 1954).

The fission product inventory in the reactor at the time of the destructive test was calculated using the RSAC-4 computer code. Based on the information contained in operations reports (Dietrich and Layman, 1954; Dietrich, 1954) the reactor was assumed to have operated for a total of 25,000 MW-s in 1953 and 280 MW-s during the early summer of 1954. To estimate the quantity of fission products that were released during this test, the SPERT-I, Tests No. 1, 2, and 3 were reviewed for similarities. Reactor periods and energy released for the three SPERT-I tests and BORAX-I Excursion are given in Table A-34.

Table A-34. Periods and energy released during BORAX-I Excursion and SPERT-I Tests

Test	Period (ms)	Energy (MW-s)	Brief Description
BORAX-I Excursion	2.6	135	"A column of what appeared to be dark grey smoke was ejected from the reactor to a height of 80 feet."
SPERT-I, Test No. 1	3.2	30.7	Destroyed the core and ejected water and core components from the core location much the same as the BORAX-I Excursion
SPERT-I, Test No. 2	2.2	155	Ruptured two rods
SPERT-I, Test No. 3	1.55	165	Ruptured two rods

Based on the information in this table, the BORAX-I Excursion and SPERT-I, Test No. 1 shared the greatest similarities. Both of these tests were fueled with plate type fuel elements. For SPERT-I, Test No. 1, the release fractions to the atmosphere were estimated to be 7.2% of the total test-generated noble gases and less than 0.01% of the halogens (Bunch, 1965).

The energy released by the SPERT-I, Test No. 1 was about one-fourth of the energy released by the BORAX-I Excursion. The assumed release fraction for noble gases was increased by a factor of 4, to 30% for the BORAX-I Excursion. The SPERT-I, Test No. 1 release fraction for halogens was also multiplied by a factor of 4 and an additional factor of 10 to ensure a conservative estimate. Therefore, the assumed halogen release fraction for the BORAX-I Excursion was estimated to be 0.4%. Based on engineering judgment, the release fractions for fission products solids and uranium were assumed to be 0.1%, one-fourth of the release fraction for the halogens.

A.3 RESULTS

A.3.1 Operational and Episodic Release Source Term Tables

Tables A-35 through A-40 present the airborne effluent releases from INEL facility from 1989 through 1952. Source terms for episodic releases except the CERTS are shown in Table A-41.

Table A-35. Airborne effluent releases from INEL facilities (1989 to 1983)

Radionuclide	Annual Activity Released (Ci) 1989	1988	1987	1986	1985	1984	1983
H-3	2.7E+00	7.0E+02	9.0E+02	6.2E+01	2.2E+02	3.6E+02	6.2E+02
C-14	2.1E-01	2.7E+00	4.0E+00	6.1E-01	7.0E-01	3.3E-01	2.3E-01
N-13							
Na-24	2.7E-03	8.9E-03	1.7E-03	1.0E-03			
Ar-41	1.4E+03	2.1E+03	2.5E+03	1.8E+03	2.1E+03	1.8E+03	2.3E+03
Sc-46							
Cr-51	2.3E-03	1.0E-02	5.8E-03	1.2E-02	1.3E-02		
Mn-54							
Mn-56							
Co-58							
Co-60			1.2E-02				
As-76							
Br-82	4.9E-03	2.2E-02	8.3E-03	1.0E-02	1.6E-02	4.6E-02	8.7E-01
Kr-85	2.0E+04	1.7E+05	2.5E+05	1.1E+04	6.4E+04	6.7E+01	3.1E+03
Kr-85m	3.6E+01	3.8E+01	4.4E+01	7.1E+01	1.8E+02	1.2E+02	2.5E+02
Kr-87	7.6E+01	7.7E+01	1.1E+02	1.5E+02	2.1E+02	3.4E+02	8.6E+02
Kr-88	7.9E+01	7.4E+01	1.2E+02	1.6E+02	3.6E+02	3.7E+02	8.8E+02
Kr-89							
Kr-90							
Rb-89	2.6E-01	8.2E-01	2.1E-01	2.3E-03			
Rb-90							
Sr-89							
Sr-90	9.1E-05	3.4E-04	2.6E-04	2.0E-04	1.9E-03	1.6E-03	1.3E-03
Sr-91							
Y-91							
Y-91m	1.5E-03	1.8E-03					
Zr-95							
Nb-95							
Tc-99m		3.7E-03					
Ru-103							
Ru-106	2.0E-03	1.9E-01	3.7E-04			3.9E-03	
Sb-125	3.9E-04	7.4E+00	1.6E+01	9.8E-01	6.8E-03	1.9E-02	2.9E-03
Te-132							
I-129(ELE)	4.7E-04	7.3E-02	6.6E-02	2.2E-03	2.6E-03	4.0E-02	4.6E-02
I-129(ORG)	9.3E-04	1.5E-01	1.3E-01	4.6E-03	5.3E-03	8.0E-02	9.4E-02
I-131	1.2E-04	1.7E-04	5.3E-04	1.1E-03	2.6E-02	1.2E-03	
I-132	3.8E-04	2.0E-03			1.0E-03		
I-133	3.5E-04	5.8E-04			4.7E-02		
I-134							
I-135					1.9E-02		
Xe-131m					8.9E+00		
Xe-133	4.0E+02	4.5E+02	6.1E+02	5.2E+02	8.2E+03	2.0E+02	3.3E+02
Xe-133m					1.9E+01	2.4E-01	
Xe-135	1.6E+02	1.9E+02	2.2E+02	4.1E+02	1.2E+03	4.0E+02	8.6E+02
Xe-135m	4.6E+01	4.3E+01	6.0E+01	8.5E+01	1.3E+02	1.7E+02	4.2E+02
Xe-137							
Xe-138	1.6E+02	2.0E+02	3.4E+02	4.1E+02	5.0E+02	8.8E+02	2.3E+03
Xe-139							
Cs-134							
Cs-136							
Cs-137	2.1E-04	6.8E-04	5.5E-04	2.4E-03	6.2E-03	8.4E-03	4.4E-03
Cs-138	4.2E-01	8.3E-01	1.9E-01	9.3E-01	2.5E+00	6.1E+00	1.6E+01
Cs-139							
Ba-139	3.0E-02	8.4E-02	1.7E-02	7.5E+00	2.7E+01	6.8E+01	1.5E+02
Ba-140							
Ce-141							
Ce-144							
Pr-143							
Pm-147							
Eu-154							
Eu-155							
Ta-182							
Ir-192							
Hg-203							
Th-232							
Np-237							
U-238+D	1.0E-05	1.3E-05					
Pu-238	7.1E-08	1.4E-05	1.7E-05	1.6E-05	6.6E-05	2.3E-04	1.4E-03
Pu-239				3.0E-07			
Pu-240				2.9E-07			
Pu-239/240	1.0E-08	2.1E-06	2.6E-06	9.0E-07	1.3E-05	9.2E-05	2.0E-04
TOTAL ACTIVITY	2.24E+04	1.74E+05	2.55E+05	1.47E+04	7.72E+04	4.78E+03	1.21E+04

Table A-36. Airborne effluent releases from INEL facilities (1982 to 1976)

	Annual Activity Released (Ci)						
Radionuclide	1982	1981	1980	1979	1978	1977	1976
H-3	2.4E+02	4.0E+02	1.8E+03	1.4E+02	2.6E+03	3.1E+03	1.0E+03
C-14	2.9E-01	1.6E+00	4.3E+00	1.1E+00	9.1E+00		
N-13							
Na-24							
Ar-41	2.5E+03	2.5E+03	2.2E+03	3.4E+03	3.8E+03	3.4E+03	5.0E+03
Sc-46							
Cr-51							
Mn-54							
Mn-56							
Co-58							
Co-60							
As-76							
Br-82	1.2E-01	9.0E-02	6.4E-02	1.1E-01	8.9E-02	1.2E-01	1.2E-01
Kr-85	8.7E+03	5.8E+04	9.0E+04	1.4E+01	9.2E+04	1.1E+05	3.2E+04
Kr-85m	1.0E+03	2.2E+02	3.6E+02	5.5E+02	5.3E+02	1.1E+03	2.9E+03
Kr-87	1.7E+03	8.6E+02	1.3E+03	1.9E+03	2.1E+03	4.8E+03	1.2E+04
Kr-88	1.7E+03	8.1E+02	1.2E+03	1.7E+03	1.8E+03	3.4E+03	9.3E+03
Kr-89					3.0E+03		
Kr-90					1.4E+02		
Rb-89							
Rb-90							
Sr-89							
Sr-90	3.5E-03	4.1E-03	5.3E-03	1.1E-01	2.4E-02	6.2E-02	4.8E-02
Sr-91				1.5E-03			
Y-91							
Y-91m							
Zr-95						1.7E-03	2.2E-03
Nb-95							
Tc-99m							
Ru-103							
Ru-106	5.5E-03	7.6E-02	7.8E-03	1.6E-02	7.0E-02	1.3E-01	1.0E-01
Sb-125	7.8E-03	1.9E-01	1.3E+00	2.5E-02	2.1E-01	3.4E-02	7.3E-02
Te-132							
I-129(ELE)	3.2E-02	1.2E-02	2.5E-02	2.1E-02	5.0E-02	3.7E-02	3.7E-02
I-129(ORG)	6.5E-02	2.4E-02	5.1E-02	4.4E-02	1.0E-01	7.3E-02	7.3E-02
I-131	5.8E-04	4.7E-02	1.8E-02	1.2E-03	2.5E-02	1.7E-03	4.0E-03
I-132					1.2E-03		
I-133					1.3E-03		1.5E-04
I-134					1.2E-01		
I-135					7.8E-02		
Xe-131m							
Xe-133	1.0E+03	1.6E+02	5.9E+02	8.8E+02	5.2E+02	6.6E+02	1.3E+03
Xe-133m				1.6E-01	5.9E-03	1.5E-01	
Xe-135	1.6E+03	8.0E+02	1.2E+03	1.8E+03	1.8E+03	3.8E+03	1.0E+04
Xe-135m	7.6E+02	4.2E+02	6.3E+02	9.9E+02	1.1E+03	2.5E+03	5.2E+03
Xe-137							
Xe-138	4.6E+03	2.5E+03	3.7E+03	5.2E+03	7.6E+03	1.0E+04	2.1E+04
Xe-139					3.8E+02		
Cs-134		6.2E-03		1.9E-03	2.9E-03	6.3E-03	2.5E-01
Cs-136							
Cs-137	9.0E-03	8.5E-03	1.9E-02	1.3E-01	8.6E-02	2.0E-01	1.1E-01
Cs-138	3.1E+01	1.7E+01	2.5E+01	3.6E+01	3.5E+01	7.3E+01	1.5E+02
Cs-139							
Ba-139	2.9E+02	1.6E+02	2.6E+02	3.6E+02	2.7E+02	5.0E+02	1.3E+03
Ba-140				2.4E-03			
Ce-141							3.0E-01
Ce-144	1.9E-03			2.2E-03	4.7E-03	2.5E-03	6.4E-03
Pr-143							
Pm-147							
Eu-154						7.6E-03	
Eu-155							
Ta-182							
Ir-192							
Hg-203							
Th-232							
Np-237							
U-238+D							
Pu-238	1.8E-04	7.6E-05	3.8E-04	5.9E-04	9.1E-04	9.9E-04	1.0E-03
Pu-239							
Pu-240							
Pu-239/240	2.0E-05	1.4E-05	4.9E-05	6.4E-05	9.8E-05	4.2E-04	4.4E-04
TOTAL ACTIVITY	2.41E+04	6.68E+04	1.03E+05	1.70E+04	1.18E+05	1.43E+05	1.01E+05

Table A-37. Airborne effluent releases from INEL facilities (1975 to 1969)

Radionuclide	Annual Activity Released (Ci)						
	1975	1974	1973	1972	1971	1970	1969
H-3	2.4E+02	6.0E+03	3.5E+02	1.6E+03	3.3E+03	3.5E+03	3.0E+03
C-14							
N-13				6.4E-02			
Na-24						3.6E+02	
Ar-41	5.1E+03	4.1E+03	4.5E+03	7.2E+03	1.6E+04	2.8E+04	5.5E+04
Sc-46							
Cr-51							5.0E-01
Mn-54	3.1E-03	5.0E-03	1.7E-02				
Mn-56							
Co-58			1.5E-03				
Co-60		3.4E-03	2.6E-03	2.8E-03	5.1E-03		
As-76	2.2E-03						
Br-82	3.5E-02	6.7E-02	2.6E-01				
Kr-85	2.4E+04	2.5E+05	4.5E+03	4.5E+04	1.3E+05	1.4E+05	1.1E+05
Kr-85m	2.5E+03	1.3E+03					
Kr-87	1.0E+04	4.2E+03	1.8E-01	2.0E-01	1.3E+00		
Kr-88	8.0E+03	3.6E+03	2.6E+03	1.3E+03	8.0E+02	1.7E+03	1.4E+03
Kr-89		5.5E+02	2.7E+02	2.9E+02	3.6E+02	2.3E+03	3.8E+03
Kr-90							
Rb-89		8.5E+00	7.3E+00	1.2E+01	1.6E+01	8.5E+01	2.1E+02
Rb-90							
Sr-89							
Sr-90	2.4E-01	3.2E+00	1.8E+00	3.5E+00	1.4E+01	3.3E+00	4.4E+00
Sr-91							
Y-91							
Y-91m							
Zr-95		4.2E-03	2.3E-02	3.3E-01	1.7E+00	1.2E+00	1.1E-02
Nb-95	3.7E-03		1.3E-02				
Tc-99m							
Ru-103			2.3E-03	6.5E-03	1.8E-01		
Ru-106	7.9E-01	3.8E+00	3.3E+00	6.1E+00	4.0E+01	3.7E+00	4.6E+00
Sb-125	4.4E-02	6.3E+00	6.7E-02	1.2E-01	1.7E+00	1.4E+00	1.2E-01
Te-132							
I-129(ELE)	2.5E-02	2.5E-02	4.0E-02	3.3E-02	3.2E-02	1.4E-02	2.9E-02
I-129(ORG)	4.9E-02	4.9E-02	8.0E-02	6.7E-02	6.4E-02	2.9E-02	5.9E-02
I-131	5.6E-02	1.1E-01	4.0E-04	3.6E+00	8.6E+00	3.1E-04	6.0E+00
I-132							
I-133							
I-134							
I-135							
Xe-131m							
Xe-133	1.4E+03	9.4E+02	5.1E+02	6.9E+00	4.2E+00		6.0E+02
Xe-133m							
Xe-135	9.0E+03	4.1E+03	1.4E+03	4.6E+00	2.7E+00		
Xe-135m	4.1E+03	1.7E+03					
Xe-137							
Xe-138	1.6E+04	6.4E+03	2.8E+03	1.7E+03	1.4E+03	3.1E+03	4.5E+03
Xe-139							
Cs-134	3.5E-01	5.5E-01	5.0E-01	8.6E-01	5.7E+00	7.0E-01	8.0E-01
Cs-136							
Cs-137	6.0E-01	6.7E+00	5.4E+00	4.6E+00	1.5E+01	2.3E+00	4.3E+00
Cs-138	1.1E+02	4.4E+01	2.7E+01	2.6E+01	2.6E+01	5.9E+01	1.4E+02
Cs-139							
Ba-139	1.1E+03	5.1E+02					
Ba-140							
Ce-141	2.0E-01			2.8E-03			
Ce-144	1.1E-01	6.8E-01	7.1E-01	3.4E+00	3.1E+01	8.2E+00	3.5E+00
Pr-143							
Pm-147							
Eu-154	4.9E-03	2.4E-02	1.3E-02	2.5E-02			
Eu-155	1.3E-03						
Ta-182							
Ir-192							
Hg-203							
Th-232	2.0E-05						
Np-237	3.9E-05						
U-238+D							
Pu-238	1.6E-03	9.6E-03	6.9E-03	8.0E-03	2.6E-02	8.7E-03	6.2E-03
Pu-239		3.0E-04					
Pu-240							
Pu-239/240	3.1E-04	8.4E-04	1.0E-03	1.2E-03	3.8E-03	1.3E-03	9.1E-04
TOTAL ACTIVITY	8.16E+04	2.83E+05	1.70E+04	5.72E+04	1.52E+05	1.79E+05	1.79E+05

Table A-38. Airborne effluent releases from INEL facilities (1968 to 1962)

Radionuclide	Annual Activity Released (Ci)						
	1968	1967	1966	1965	1964	1963	1962
H-3	2.52E+03	3.93E+03	3.02E+03	1.05E+03	3.47E+03	4.52E+03	4.53E+00
C-14							
N-13							
Na-24	6.04E+00	9.46E+00	1.60E+01	2.49E+01	2.72E+01	2.87E+01	1.67E+01
Ar-41	5.68E+04	8.91E+04	1.50E+05	2.34E+05	2.55E+05	2.69E+05	1.57E+05
Sc-46							
Cr-51	2.09E-02	3.28E-02	5.54E-02	8.63E-02	9.44E-02	9.93E-02	5.79E-02
Mn-54				8.88E-03			
Mn-56							
Co-58							
Co-60	1.59E-01	2.49E-01	4.21E-01	1.09E+00	7.17E-01	7.54E-01	4.40E-01
As-76							
Br-82							
Kr-85	8.43E+04	1.82E+04	5.25E+04	4.69E+04	8.20E+04	8.38E+04	8.35E+02
Kr-85m	2.56E+03	3.99E+03	6.77E+03	1.05E+04	1.15E+04	1.21E+04	7.08E+03
Kr-87	9.25E+03	1.45E+04	2.45E+04	3.81E+04	4.16E+04	4.39E+04	2.56E+04
Kr-88	9.06E+03	1.42E+04	2.40E+04	3.74E+04	4.08E+04	4.30E+04	2.51E+04
Kr-89							
Kr-90							
Rb-89	7.40E+02	1.16E+03	1.96E+03	3.06E+03	3.34E+03	3.51E+03	2.05E+03
Rb-90							
Sr-89					3.73E-01	1.55E+00	2.47E+00
Sr-90	1.49E+01	2.59E+00	9.64E+00	3.35E+01	8.75E+00	3.44E+01	4.67E+00
Sr-91						6.32E-02	1.01E-01
Y-91						1.73+00	2.69E+00
Y-91m	8.27E-01	1.30E+00	2.19E+00	3.41E+00	3.73E+00	3.93E+00	2.29E+00
Zr-95	7.34E+00	1.53E-02	5.86E+00	1.27E+01	3.07E+00	1.81E+00	2.73E+00
Nb-95							
Tc-99m	1.84E+00	2.89E+00	4.88E+00	7.60E+00	8.31E+00	8.74E+00	5.10E+00
Ru-103						1.23E+00	1.97E+00
Ru-106	9.18E+00	2.14E+01	1.53E+02	2.44E+01	3.42E+02	4.64E+00	2.58E-01
Sb-125	4.86E-02		2.02E+00	9.43E-01			
Te-132		3.92E-01				2.91E+00	4.67E+00
I-129(ELE)	6.00E-02	1.30E-01	9.67E-02	3.33E-04	1.83E-01	1.83E-02	1.13E-08
I-129(ORG)	1.20E-01	2.60E-01	1.93E-01	6.67E-04	3.67E-01	3.67E-02	2.27E-08
I-131	4.15E+00	2.27E+00	5.35E+00	9.11E+00	1.39E+00	2.65E+01	4.10E+01
I-132		5.51E-02	3.59E+00			8.25E+01	1.30E+02
I-133	2.44E-01	1.28E+00	2.32E+00	1.03E+00	1.10E+00	1.01E+01	1.50E+01
I-134							
I-135		2.81E-01	4.47E-01			1.05E-01	1.69E-01
Xe-131m						7.80E+02	1.76E+03
Xe-133	1.16E+03	3.57E+03	5.15E+03	2.22E+04	4.44E+03	2.69E+05	5.97E+05
Xe-133m						5.81E+03	1.31E+04
Xe-135	9.82E+03	1.53E+04	2.60E+04	4.04E+04	5.19E+04	5.37E+04	4.34E+04
Xe-135m	5.40E+03	8.47E+03	1.43E+04	2.23E+04	2.43E+04	2.56E+04	1.50E+04
Xe-137							
Xe-138	3.03E+04	4.75E+04	8.03E+04	1.25E+05	1.36E+05	1.44E+05	8.39E+04
Xe-139							
Cs-134	6.73E-01	1.80E-01	1.87E+00	3.80E+00	9.27E-01	9.77E-01	3.36E-01
Cs-136						5.52E-02	8.85E-02
Cs-137	5.67E+00	1.38E+00	5.31E+00	1.39E+01	4.79E+00	3.06E+01	2.22E+00
Cs-138	6.68E+02	1.04E+03	1.77E+03	2.76E+03	3.01E+03	3.17E+03	1.85E+03
Cs-139							
Ba-139	6.31E+01	9.92E+01	1.67E+02	2.61E+02	2.85E+02	3.00E+02	1.75E+02
Ba-140	2.17E-02	3.40E-02	5.75E-02	8.96E-02	9.79E-02	4.88E+00	7.73E+00
Ce-141	1.38E-03	2.17E-03	3.66E-03	5.70E-03	6.23E-03	2.68E+00	4.29E+00
Ce-144	6.17E+01	8.60E-01	3.41E+01	5.67E+01	2.26E+01	4.12E+01	2.56E+00
Pr-143						4.71E+00	7.57E+00
Pm-147						1.16E+02	5.64E+00
Eu-154						5.79E-02	2.84E-03
Eu-155							
Ta-182							
Ir-192							
Hg-203	2.37E+00	3.71E+00	6.27E+00	9.76E+00	1.06E+01	1.12E+01	6.55E+00
Th-232							
Np-237							
U-238+D							
Pu-238	2.90E-02	1.50E-03	1.33E-02	5.87E-02	1.36E-03	3.51E-02	3.75E-03
Pu-239	1.55E-04	2.42E-04	4.10E-04	6.38E-04	6.98E-04	7.34E-04	4.28E-04
Pu-240							
Pu-239/240	4.20E-03	2.20E-04	1.96E-03	8.63E-03	2.00E-04	5.15E-03	5.51E-04
TOTAL ACTIVITY	2.13E+05	2.21E+05	3.91E+05	5.84E+05	6.58E+05	9.63E+05	9.74E+05

Table A-39. Airborne effluent releases from INEL facilities (1961 to 1955)

Radionuclide	Annual Activity Released (Ci)						
	1961	1960	1959	1958	1957	1956	1955
H-3	8.56E+00	6.05E+00	6.41E+03	7.81E+03	5.51E+03	2.81E+03	2.71E+03
C-14							
N-13							
Na-24	1.61E+01	1.11E+01	8.09E+00	4.96E+00	3.66E+00	9.49E+00	1.93E+01
Ar-41	1.50E+05	1.04E+05	7.58E+04	4.65E+04	3.42E+04	8.89E+04	1.81E+05
Sc-46							
Cr-51	5.56E-02	3.86E-02	2.80E-02	1.72E-02	1.27E-02	3.29E-02	6.70E-02
Mn-54							
Mn-56							
Co-58							
Co-60	4.23E-01	2.93E-01	2.13E-01	1.31E-01	9.62E-02	2.50E-01	5.09E-01
As-76							
Br-82							
Kr-85	8.82E+02	6.13E+02	1.17E+05	1.42E+05	1.00E+05	5.44E+04	5.04E+04
Kr-85m	6.82E+03	4.72E+03	3.44E+03	2.12E+03	1.56E+03	4.02E+03	8.19E+03
Kr-87	2.46E+04	1.70E+04	1.24E+04	7.59E+03	5.59E+03	1.45E+04	2.96E+04
Kr-88	2.41E+04	1.67E+04	1.21E+04	8.25E+03	5.48E+03	1.42E+04	2.90E+04
Kr-89							
Kr-90							
Rb-89	1.97E+03	1.36E+03	9.91E+02	6.08E+02	4.48E+02	1.16E+03	2.37E+03
Rb-90							
Sr-89	2.44E+00	9.38E-01	7.56E+00	1.48E+01	1.92E+01	2.07E+01	1.73E+01
Sr-90	3.15E+00	2.60E+00	4.25E+01	5.08E+01	3.57E+01	2.18E+01	2.11E+01
Sr-91	1.00E-01	3.85E-02	4.31E-01	1.02E+00	9.19E-01		
Y-91	2.66E+00	1.02E+00	8.29E+00	1.61E+01	2.11E+01	4.36E+01	3.72E+01
Y-91m	2.20E+00	1.52E+00	1.11E+00	6.79E-01	5.01E-01	1.30E+00	2.65E+00
Zr-95	2.69E+00	1.03E+00	8.47E+00	1.64E+01	2.13E+01	6.36E+01	5.45E+01
Nb-95							
Tc-99m	4.90E+00	3.39E+00	2.47E+00	1.51E+00	1.11E+00	2.89E+00	5.90E+00
Ru-103	1.95E+00	7.49E-01	6.05E+00	1.18E+01	1.53E+01	4.46E+00	3.38E+00
Ru-106	3.42E-02	8.11E-02	6.65E+00	8.18E+00	5.87E+00	2.03E+01	1.77E+01
Sb-125							
Te-132	4.61E+00	1.77E+00	1.51E+01	3.06E+01	3.41E+01	1.50E-01	
I-129(ELE)	1.19E-08	9.02E-09	6.13E-08	2.78E-07	4.30E-07	1.08E-08	
I-129(ORG)	2.37E-08	1.80E-08	1.23E-07	5.56E-07	8.60E-07	2.16E-08	
I-131	4.28E+01	3.26E+01	2.24E+02	1.03E+03	1.40E+03	1.16E+01	9.83E-01
I-132	2.27E+02	1.76E+02	1.55E+03	3.38E+03	4.03E+03	9.83E+00	
I-133	1.56E+01	1.19E+01	1.00E+02	4.84E+02	4.41E+02	3.91E-01	7.83E-01
I-134							
I-135	1.77E-01	1.35E-01	1.51E+00	7.67E+00	6.65E+00	2.08E-11	
Xe-131m	3.32E+03	2.34E+03	3.17E+03	2.41E+03	1.65E+03	3.22E+02	
Xe-133	1.12E+06	7.94E+05	1.09E+06	8.48E+05	4.63E+05	2.70E+04	3.16E+03
Xe-133m	2.47E+04	1.74E+04	2.49E+04	2.02E+04	9.16E+03	1.16E+02	
Xe-135	5.67E+04	3.98E+04	5.12E+04	4.54E+04	1.83E+04	1.54E+04	3.14E+04
Xe-135m	1.44E+04	9.97E+03	7.25E+03	4.46E+03	3.28E+03	8.49E+03	1.73E+04
Xe-137							
Xe-138	8.06E+04	5.59E+04	4.06E+04	2.49E+04	1.83E+04	4.76E+04	9.71E+04
Xe-139							
Cs-134	3.03E-01	2.16E-01	7.76E-01	8.55E-01	6.07E-01	9.75E-01	1.06E+00
Cs-136	9.27E-02	7.06E-02	4.97E-01	2.23E+00	3.17E+00	3.69E-02	
Cs-137	7.67E-01	9.58E-01	4.21E+01	5.09E+01	3.59E+01	2.06E+01	1.85E+01
Cs-138	1.78E+03	1.23E+03	8.95E+02	5.49E+02	4.04E+02	1.05E+03	2.14E+03
Cs-139							
Ba-139	1.68E+02	1.16E+02	8.46E+01	5.19E+01	3.82E+01	9.92E+01	2.02E+02
Ba-140	7.63E+00	2.95E+00	2.37E+01	4.67E+01	5.85E+01	1.34E+00	6.95E-02
Ce-141	4.24E+00	1.63E+00	1.31E+01	2.57E+01	3.32E+01	3.37E+00	1.91E+00
Ce-144	5.70E-01	8.21E-01	5.97E+01	7.39E+01	5.40E+01	2.77E+02	2.41E+02
Pr-143	7.47E+00	2.87E+00	2.32E+01	4.56E+01	5.79E+01	1.40E+00	
Pm-147	4.04E-02	1.72E+00	1.64E+02	2.00E+02	1.40E+02	6.69E+01	5.82E+01
Eu-154			8.21E-02	1.00E-01	7.05E-02	4.81E-02	
Eu-155							
Ta-182							
Ir-192							
Hg-203	6.29E+00	4.36E+00	3.17E+00	1.94E+00	1.43E+00	3.72E+00	7.58E+00
Th-232							
Np-237							
U-238+D							
Pu-238	5.69E-03	3.12E-02	9.07E-02	1.12E-01	7.78E-02	4.44E-02	2.75E-02
Pu-239	4.11E-04	2.85E-04	2.07E-04	1.27E-04	9.36E-05	2.43E-04	4.95E-04
Pu-240							
Pu-239/240	8.37E-04	4.55E-03	1.33E-02	1.64E-02	1.15E-02	6.53E-03	4.04E-03
TOTAL ACTIVITY	1.51E+06	1.07E+06	1.45E+06	1.17E+06	6.73E+05	2.81E+05	4.55E+05

Table A-40. Airborne effluent releases from INEL facilities (1954 to 1952)

Radionuclide	Annual Activity Released (Ci) 1954	1953	1952
H-3	1.93E+03	7.75E+02	
C-14			
N-13			
Na-24	1.30E+01	1.77E+01	8.05E+00
Ar-41	1.22E+05	1.66E+05	7.53E+04
Sc-46			
Cr-51	4.50E-02	6.13E-02	2.79E-02
Mn-54			
Mn-56			
Co-58			
Co-60	3.42E-01	4.66E-01	2.12E-01
As-76			
Br-82			
Kr-85	3.58E+04	1.49E+04	3.60E+02
Kr-85m	5.50E+03	7.49E+03	3.40E+03
Kr-87	1.99E+04	2.71E+04	1.23E+04
Kr-88	1.95E+04	2.65E+04	1.21E+04
Kr-89			
Kr-90			
Rb-89	1.59E+03	2.17E+03	9.86E+02
Rb-90			
Sr-89	1.23E+01	4.92E+00	
Sr-90	1.48E+01	8.37E+00	1.57E+00
Sr-91			
Y-91	2.65E+01	1.06E+01	
Y-91m	1.78E+00	2.42E+00	1.10E+00
Zr-95	3.87E+01	1.55E+01	
Nb-95			
Tc-99m	3.96E+00	5.40E+00	2.45E+00
Ru-103	2.40E+00	9.61E-01	
Ru-106	1.26E+01	5.03E+00	
Sb-125			
Te-132			
I-129(ELE)			
I-129(ORG)			
I-131	6.61E-01	8.99E-01	4.09E-01
I-132			
I-133	5.26E-01	7.16E-01	3.26E-01
I-134			
I-135			
Xe-131m			
Xe-133	2.12E+03	2.89E+03	1.31E+03
Xe-133m			
Xe-135	2.11E+04	2.87E+04	1.30E+04
Xe-135m	1.16E+04	1.58E+04	7.19E+03
Xe-137			
Xe-138	6.52E+04	8.88E+04	4.04E+04
Xe-139			
Cs-134	7.37E-01	5.29E-01	1.51E-01
Cs-136			
Cs-137	1.31E+01	5.82E+00	3.75E-01
Cs-138	1.44E+03	1.96E+03	8.90E+02
Cs-139			
Ba-139	1.36E+02	1.85E+02	8.41E+01
Ba-140	4.67E-02	6.36E-02	2.89E-02
Ce-141	1.36E+00	5.48E-01	1.84E-03
Ce-144	1.72E+02	6.86E+01	
Pr-143			
Pm-147	4.14E+01	1.65E+01	
Eu-154		1.19E-02	
Eu-155			
Ta-182			
Ir-192			
Hg-203	5.09E+00	6.93E+00	3.15E+00
Th-232			
Np-237			
U-238+D			
Pu-238	2.75E-02	1.10E-02	
Pu-239	3.33E-04	4.53E-04	2.06E-04
Pu-240			
Pu-239/240	4.04E-03	1.62E-03	
TOTAL ACTIVITY	3.08E+05	3.83E+05	1.67E+05

Table A-41. Episodic release source terms

Radionuclide	SNAPTRAN-2 1/11/66	SPERT-I, Test No. 3 4/14/64	SNAPTRAN-3 4/1/64	SPERT-I, Test No. 2 11/10/63	SPERT-I, Test No. 1 11/5/62	1961 ICPP Criticality 1/25/61	SL-1 Accident 1/3/61
Ar-41	1.59E+03		4.78E+03				
Br-84	2.06E-01	1.14E-03	2.26E-02	5.21E-06	2.78E-04	1.28E-02	1.90E-02
Kr-85m	1.96E+00	5.49E-03	5.88E-02	5.66E-04	1.31E-02	2.80E-01	3.55E+01
Kr-87	2.57E+01	1.68E-01	2.40E+00	6.04E-03	7.41E-01	6.06E+00	2.52E+02
Kr-88+D	3.91E+01	1.52E-01	2.08E+00	1.23E-02	1.47E+00	1.08E+01	2.14E+02
Rb-89	5.88E-01	1.44E-01	1.04E+01	1.56E-05	3.45E+00	8.21E-01	5.46E-02
Sr-89	2.44E-01	5.84E-04	6.66E-03	9.60E-05	1.05E-02	8.49E-02	3.25E+00
Sr-90+D	1.25E-03	3.03E-06	5.50E-05	4.74E-07	9.40E-05	5.81E-04	1.00E-01
Sr-91+D	1.99E+01	5.30E-02	1.01E+00	6.81E-03	1.65E+00	9.06E+00	4.13E-03
Sr-92	2.00E+01	6.51E-02	1.38E+00	5.10E-03	2.07E+00	7.77E+00	1.21E-02
Y-91	2.48E-02	2.04E-05	1.58E-04	1.26E-05	5.75E-04	8.69E-03	1.95E+01
Y-92	1.24E+01	1.58E-02	1.64E-01	4.39E-03	4.60E-01	3.89E+00	1.19E-02
Y-93	4.41E+00	7.48E-03	1.37E-01	9.74E-04	2.34E-01	1.03E+00	2.06E-02
Zr-95+D	1.47E-02				9.83E-06		2.13E+01
Zr-97	1.09E+00				1.67E-04		3.28E-02
Nb-96	8.13E-05				1.22E-08		1.51E-05
Mo-99	3.05E-01				4.62E-05		3.66E+00
Ru-103+D	1.17E-02				5.34E-06		1.01E+01
Ru-105	5.37E-01				1.01E-04		7.06E-03
Ru-106+D	1.66E-04				2.43E-07		7.17E-01
Sb-129	3.52E-01				6.63E-05		4.67E-03
Te-131	8.51E-01				8.77E-04		7.18E-02
Te-131m	2.53E-01				5.54E-06		2.74E-02
Te-132+D	8.79E-01				2.92E-05		3.49E+00
Te-133m	1.40E+01				6.93E-04		5.18E-02
Te-134	1.96E+01				1.55E-03		1.18E-01
I-131 (ELE)	6.93E-02	3.44E-06	3.14E-05	5.34E-07	3.93E-05	1.01E-01	8.00E+01
I-131 (ORG)							
I-132	7.46E-01	1.00E-03	1.09E-02	6.96E-05	5.60E-05	1.88E-01	4.95E+01
I-133	7.62E+00	1.23E-03	1.14E-02	1.75E-04	1.57E-04	2.02E+00	1.33E+00
I-134	6.47E+01	3.31E-02	4.84E-01	6.14E-04	2.62E-03	3.85E+00	1.54E+00
I-135	2.17E+01	6.39E-02	6.19E-01	7.54E-03	2.23E-03	5.74E+00	4.07E-01
Xe-129m	1.78E-09	4.37E-12	7.96E-11	6.78E-13	2.06E-10	8.44E-10	7.65E-05
Xe-135	6.49E+00	9.94E-03	9.87E-02	2.90E-03	1.33E-01	1.83E+00	2.09E+01
Xe-135m	3.74E+00	1.46E-02	4.66E-01	1.29E-03	1.69E-01	1.02E+00	3.90E+01
Xe-138	3.36E-01	1.24E-01	1.15E+01	6.89E-06	3.79E+00	5.99E-01	3.01E+02
Cs-137+D	1.47E-03	3.73E-06	4.96E-05	5.83E-07	5.43E-05	4.38E-04	5.00E-01
Cs-138	3.13E+01	6.08E-01	1.52E+01	3.37E-03	1.53E+01	2.35E+01	1.63E-01
Ba-139	6.46E+01	3.65E-01	7.53E+00	1.49E-02	1.01E+01	3.54E+01	1.08E-01
Ba-140+D	8.01E-01	1.89E-03	3.44E-02	2.94E-04	5.86E-02	3.57E-01	1.56E+01
Ba-141	4.29E-01	4.16E-02	2.96E+00	1.39E-05	1.62E+00	4.48E-01	6.98E-02
Ba-142	3.15E-03	3.26E-03	6.10E-01	1.41E-08	1.49E-01	4.84E-03	2.30E-02
La-141	1.55E+01	3.88E-02	6.07E-01	4.12E-03	1.20E+00	5.45E+00	4.13E-02
La-142	8.87E+00	2.54E-02	5.32E-01	1.24E-03	8.17E-01	2.28E+00	8.99E-02
Ce-141	4.20E-02	3.07E-05	1.98E-04	1.72E-05	8.53E-04	1.11E-02	1.86E+01
Ce-143	5.97E-01				8.48E-05		6.30E-01
Ce-144+D	2.78E-03				3.86E-06		1.10E+01
Pr-143	3.33E-03				8.07E-06		1.66E+01
Pr-144	2.78E-03				3.85E-06		1.10E+01
U-234	1.21E-02				2.41E-06		
U-235	3.84E-04				7.68E-08		
U-238	3.57E-06				7.14E-10		

Table A-41. (continued)

Radionuclide	IET #26(B) 3/1-30/61	IET #26(A) 12/23-2/28/61	IET #25(B) 12/1-15/60	IET #25(A) 11/22-30/60	IET #24 (LIME-II) 10/26/60	IET #23 (FEET #2) 9/7-10/14/60	IET #22 (LIME) 8/25/60
Ar-41	2.59E+03	6.51E+03	7.62E+03	2.38E+03	6.52E+01	7.24E+02	2.22E+01
Br-84	1.85E+00	1.06E+00	9.24E-02	6.58E-02	3.17E+01	1.15E+01	1.76E+01
Kr-85m	3.13E+01	3.14E+01	3.52E+00	1.56E+00	1.33E+02	3.38E+01	2.79E+01
Kr-87	7.77E+01	6.58E+01	2.64E+00	1.61E+00	8.76E+02	1.42E+02	3.44E+02
Kr-88+D	1.42E+02	1.33E+02	1.08E+01	5.30E+00	8.14E+02	1.77E+02	2.82E+02
Rb-89	4.70E+00	4.46E+00	4.40E-03	1.68E-03	2.63E+02	3.41E+01	6.26E+02
Sr-89	3.98E-02	2.91E-02	2.94E-03	1.11E-03	7.70E-02	4.64E-02	2.96E-01
Sr-90+D	2.50E-04	1.23E-04	8.64E-06	1.68E-06	1.17E-04	2.73E-04	5.30E-04
Sr-91+D	1.45E-01	1.48E-01	2.11E-02	8.70E-03	6.82E-01	2.86E-01	2.64E+00
Sr-92	1.32E-01	1.21E-01	9.49E-03	4.78E-03	5.15E-01	3.30E-01	2.80E+00
Y-91	2.67E-02	1.51E-02	1.07E-03	1.20E-04	3.64E-04	2.41E-02	2.10E-02
Y-92	1.15E-01	1.26E-01	1.80E-02	7.64E-03	1.06E-01	1.82E-01	3.18E-01
Y-93	8.88E-02	9.91E-02	1.68E-02	6.03E-03	1.39E-01	1.73E-01	7.75E-01
Zr-95+D	2.81E-02	1.60E-02	1.18E-03	1.54E-04	1.17E-03	2.63E-02	2.54E-02
Zr-97	6.50E-02	7.60E-02	1.51E-02	4.76E-03	7.67E-02	1.21E-01	5.37E-01
Nb-96	5.71E-06	6.76E-06	1.40E-06	4.14E-07	5.75E-06	1.02E-05	4.87E-05
Mo-99	4.43E-02	5.21E-02	9.48E-03	2.25E-03	2.30E-02	6.40E-02	3.39E-01
Ru-103+D	1.78E-02	1.12E-02	8.91E-04	1.21E-04	9.47E-04	1.58E-02	2.03E-02
Ru-105	1.99E-02	1.99E-02	2.19E-03	9.70E-04	4.28E-02	4.32E-02	2.46E-01
Ru-106+D	4.43E-04	2.15E-04	1.42E-05	1.76E-06	1.35E-05	4.53E-04	2.97E-04
Sb-129	1.30E-02	1.30E-02	1.42E-03	6.30E-04	2.85E-02	2.85E-02	1.69E-01
Te-131	2.61E-02	2.37E-02	3.86E-04	1.98E-04	3.12E-01	1.58E-01	2.75E+00
Te-131m	3.26E-03	3.89E-03	8.07E-04	2.27E-04	2.70E-03	5.55E-03	2.60E-02
Te-132+D	3.14E-02	3.66E-02	6.36E-03	1.45E-03	1.43E-02	4.32E-02	2.23E-01
Te-133m	3.47E-02	2.94E-02	7.54E-04	5.04E-04	2.75E-01	1.59E-01	2.44E+00
Te-134	5.63E-02	4.89E-02	7.66E-04	5.45E-04	6.11E-01	3.26E-01	6.17E+00
I-131 (ELE)	5.01E+00	4.72E+00	5.10E+00	8.44E-01	7.84E-01	4.55E+00	1.67E+00
I-131 (ORG)							
I-132	4.38E+00	4.00E+00	3.84E+00	9.64E-01	2.14E+00	5.56E+00	4.20E+00
I-133	1.57E+01	1.88E+01	3.18E+01	9.35E+00	1.82E+01	2.27E+01	7.05E+00
I-134	2.11E+01	1.35E+01	2.94E+00	1.90E+00	1.62E+02	6.24E+01	2.58E+01
I-135	2.56E+01	2.69E+01	3.03E+01	1.21E+01	6.74E+01	4.48E+01	2.89E+01
Xe-129m	8.24E-08	8.26E-08	1.02E-08	1.76E-09	3.01E-08	4.02E-08	3.22E-08
Xe-135	9.31E+01	1.23E+02	3.95E+01	1.18E+01	8.77E+01	7.89E+01	2.82E+01
Xe-135m	6.54E+00	7.19E+00	5.20E+00	2.08E+00	8.06E+01	1.54E+01	4.20E+01
Xe-138	2.24E+01	1.98E+01	1.37E-02	4.25E-03	1.26E+03	1.60E+02	1.26E+03
Cs-137+D	3.63E-04	2.28E-04	2.15E-05	8.37E-06	9.73E-04	4.83E-04	2.93E-03
Cs-138	4.35E+01	3.92E+01	4.90E-01	3.65E-01	9.21E+02	1.32E+02	1.26E+03
Ba-139	1.60E+00	1.35E+00	6.45E-02	4.21E-02	1.98E+01	3.80E+00	7.30E+01
Ba-140+D	4.84E-02	4.36E-02	4.74E-03	9.00E-04	3.50E-02	4.64E-02	2.16E-01
Ba-141	2.63E-02	2.45E-02	4.32E-05	2.15E-05	8.94E-01	2.61E-01	7.61E+00
Ba-142	8.47E-03	6.57E-03	1.35E-06	1.89E-07	3.02E-01	1.22E-01	4.30E+00
La-141	1.30E-01	1.26E-01	1.29E-02	5.97E-03	3.07E-01	2.74E-01	1.43E+00
La-142	1.18E-01	1.01E-01	5.16E-03	2.94E-03	5.37E-01	3.63E-01	3.45E+00
Ce-141	3.55E-02	2.33E-02	1.92E-03	2.46E-04	7.25E-04	3.01E-02	3.73E-02
Ce-143	5.16E-02	6.15E-02	1.27E-02	3.51E-03	4.09E-02	8.65E-02	4.13E-01
Ce-144+D	7.29E-03	3.57E-03	2.37E-04	2.96E-05	2.27E-04	7.41E-03	4.99E-03
Pr-143	4.11E-02	3.44E-02	2.95E-03	2.91E-04	8.08E-04	3.11E-02	6.38E-02
Pr-144	7.29E-03	3.57E-03	2.37E-04	2.96E-05	2.20E-04	7.40E-03	4.88E-03
U-234	2.58E-05	6.54E-05	1.17E-05	3.65E-06	9.16E-07	1.97E-05	5.52E-05
U-235	8.20E-07	2.08E-06	3.69E-07	1.16E-07	2.92E-08	6.26E-07	1.76E-06
U-238	7.63E-09	1.94E-08	3.43E-09	1.08E-09	2.71E-10	5.82E-09	1.64E-08

Table A-41. (continued)

Radionuclide	IET #21 6/29-8/6/60	IET #20 5/14-6/10/60	IET #19(B) 3/1-4/30/60	IET #19(A) 2/17-29/60	IET #18 1/6-2/5/60	IET #17(B) 12/1-12/12/59	IET #17(A) 11/2-30/59
Ar-41	9.75E+02	5.13E+03	7.44E+03	1.03E+03	1.36E+04	2.01E+03	2.17E+03
Br-84	1.47E+00	4.34E+00	4.02E+00	2.76E-01	2.20E-01	1.75E-02	7.47E-02
Kr-85m	5.41E+01	1.40E+02	4.92E+01	1.29E+01	5.20E+00	1.18E+01	1.52E+01
Kr-87	2.01E+02	3.02E+02	1.19E+02	1.84E+01	2.96E+00	1.00E+01	1.83E+01
Kr-88+D	2.77E+02	5.70E+02	2.11E+02	4.94E+01	1.31E+01	3.96E+01	4.89E+01
Rb-89	1.83E+01	2.62E+01	9.22E+00	1.99E-01	3.09E-02	5.61E-03	8.27E-01
Sr-89	5.62E-02	7.15E-02	3.54E-02	1.07E-02	6.68E-03	1.58E-02	1.81E-02
Sr-90+D	1.66E-04	2.33E-04	1.54E-04	2.14E-05	1.78E-05	5.13E-05	5.57E-05
Sr-91+D	4.20E-01	6.27E-01	2.23E-01	6.76E-02	3.85E-02	7.17E-02	1.20E-01
Sr-92	5.06E-01	5.13E-01	1.92E-01	4.45E-02	1.12E-02	3.50E-02	4.78E-02
Y-91	1.88E-02	2.91E-02	1.83E-02	2.10E-03	2.24E-03	6.46E-03	7.09E-03
Y-92	3.11E-01	5.57E-01	1.85E-01	5.92E-02	2.83E-02	6.04E-02	7.85E-02
Y-93	2.76E-01	4.80E-01	1.59E-01	4.38E-02	3.12E-02	5.00E-02	7.91E-02
Zr-95+D	2.08E-02	3.23E-02	1.96E-02	2.35E-03	2.40E-03	6.76E-03	7.49E-03
Zr-97	1.92E-01	3.77E-01	1.26E-01	3.33E-02	3.09E-02	4.39E-02	7.25E-02
Nb-96	1.63E-05	3.35E-05	1.15E-05	2.88E-06	3.02E-06	4.15E-06	6.85E-06
Mo-99	1.08E-01	2.38E-01	8.66E-02	1.81E-02	2.26E-02	3.25E-02	5.41E-02
Ru-103+D	1.53E-02	2.45E-02	1.37E-02	1.77E-03	1.83E-03	4.94E-03	5.69E-03
Ru-105	6.97E-02	8.81E-02	3.10E-02	8.12E-03	3.18E-03	7.43E-03	9.54E-03
Ru-106+D	2.61E-04	3.87E-04	2.67E-04	2.82E-05	2.82E-05	8.52E-05	8.95E-05
Sb-129	4.57E-02	5.75E-02	2.02E-02	5.29E-03	2.06E-03	4.83E-03	6.20E-03
Te-131	1.91E-01	1.26E-01	5.04E-02	4.13E-03	1.04E-03	1.19E-03	6.75E-03
Te-131m	8.96E-03	1.90E-02	6.63E-03	1.60E-03	1.80E-03	2.45E-03	4.05E-03
Te-132+D	7.43E-02	1.63E-01	5.96E-02	1.22E-02	1.52E-02	2.25E-02	3.72E-02
Te-133m	2.06E-01	1.43E-01	5.69E-02	6.83E-03	1.07E-03	2.84E-03	7.74E-03
Te-134	3.76E-01	2.51E-01	9.98E-02	9.29E-03	1.51E-03	2.75E-03	1.24E-02
I-131 (ELE)	1.67E+00	8.49E+00	1.10E+01	1.56E+00	2.17E+01	1.45E+00	7.69E-01
I-131 (ORG)							
I-132	1.76E+00	1.36E+01	1.44E+01	1.22E+00	5.23E+00	6.45E-01	4.69E-01
I-133	7.16E+00	5.11E+01	5.07E+01	1.06E+01	1.21E+02	6.11E+00	3.91E+00
I-134	1.44E+01	4.72E+01	4.57E+01	5.06E+00	3.94E+00	6.78E-01	8.99E-01
I-135	1.36E+01	6.86E+01	6.67E+01	1.59E+01	8.18E+01	6.30E+00	4.46E+00
Xe-129m	7.18E-08	2.80E-07	1.13E-07	2.01E-08	2.33E-08	4.50E-08	6.53E-08
Xe-135	1.19E+02	7.51E+02	2.33E+02	5.12E+01	1.09E+02	6.65E+01	1.44E+02
Xe-135m	8.32E+00	3.08E+01	1.66E+01	2.84E+00	1.41E+01	1.08E+00	1.22E+00
Xe-138	7.94E+01	1.18E+02	4.03E+01	7.01E-01	1.05E-01	1.41E-02	3.48E+00
Cs-137+D	4.37E-04	5.44E-04	2.87E-04	7.81E-05	4.95E-05	1.15E-04	1.28E-04
Cs-138	1.74E+02	2.03E+02	8.43E+01	7.63E+00	1.28E+00	1.59E+00	1.09E+01
Ba-139	5.35E+00	5.83E+00	2.40E+00	4.10E-01	6.64E-02	2.39E-01	4.55E-01
Ba-140+D	7.42E-02	1.30E-01	5.63E-02	1.05E-02	1.05E-02	2.28E-02	3.12E-02
Ba-141	1.65E-01	1.42E-01	5.18E-02	1.59E-03	2.54E-04	7.89E-05	5.00E-03
Ba-142	4.46E-02	3.99E-02	1.24E-02	9.33E-05	1.26E-05	5.26E-07	9.43E-04
La-141	4.52E-01	5.43E-01	1.95E-01	5.12E-02	1.79E-02	4.53E-02	5.98E-02
La-142	5.50E-01	4.46E-01	1.74E-01	3.13E-02	5.42E-03	1.97E-02	3.05E-02
Ce-141	3.12E-02	5.20E-02	2.85E-02	3.71E-03	4.03E-03	1.06E-02	1.24E-02
Ce-143	1.40E-01	3.00E-01	1.05E-01	2.49E-02	2.87E-02	4.01E-02	6.45E-02
Ce-144+D	4.35E-03	6.48E-03	4.43E-03	4.73E-04	4.73E-04	1.42E-03	1.50E-03
Pr-143	5.04E-02	8.72E-02	4.13E-02	5.87E-03	6.34E-03	1.67E-02	2.08E-02
Pr-144	4.35E-03	6.48E-03	4.43E-03	4.73E-04	4.73E-04	1.42E-03	1.50E-03
U-234	1.97E-05	3.87E-05	3.16E-06	6.54E-07	2.10E-05	1.40E-05	1.67E-05
U-235	6.26E-07	1.23E-06	1.00E-07	2.08E-08	6.67E-07	4.46E-07	5.30E-07
U-238	5.82E-09	1.15E-08	9.34E-10	1.94E-10	6.20E-09	4.15E-09	4.93E-09

Table A-41. (continued)

Radionuclide	1959 ICPP Criticality 10/16/59	IET #16 10/9/59	IET #15(B) 6/16-24/59	IET #15(A) 6/3-15/59	IET #14 4/24-5/19/59	IET #13 (HTRE No. 3 Excursion) 11/18/58	FECF Filter Break 10/29-30/58
Ar-41		2.66E+02	9.95E+02	1.73E+03	3.80E+03	6.00E-01	
Br-84	4.93E+00	3.07E-03	1.66E+00	4.44E-01	1.12E+01	6.53E-01	
Kr-85m	1.62E+01	2.57E+00	1.12E+01	1.24E+01	2.14E+02	5.31E+00	
Kr-87	7.82E+02	5.33E+00	2.61E+01	3.95E+01	5.18E+02	5.94E+01	
Kr-88+D	9.39E+02	1.11E+01	4.81E+01	5.91E+01	9.22E+02	8.13E+01	
Rb-89	6.56E+03	8.12E-02	2.40E+00	5.19E+00	4.83E+01	5.26E+01	
Sr-89	3.95E+00	1.07E-03	6.12E-03	4.65E-03	1.47E-01	3.75E-01	1.03E-01
Sr-90+D	3.87E-02	1.24E-06	1.99E-05	1.07E-05	5.39E-04	1.99E-03	1.03E-01
Sr-91+D	6.90E+02	1.05E-02	4.74E-02	5.09E-02	1.57E+00	2.65E+01	
Sr-92	8.22E+02	1.00E-02	4.38E-02	5.46E-02	1.60E+00	1.24E+01	
Y-91	1.05E-01	2.47E-05	2.44E-03	9.51E-04	6.92E-02	1.27E-02	2.22E-01
Y-92	9.51E+01	1.02E-02	4.26E-02	3.97E-02	1.60E+00	3.63E+00	
Y-93	7.50E+01	7.04E-03	3.32E-02	3.40E-02	1.33E+00	1.31E+00	
Zr-95+D		6.98E-05	2.67E-03	1.18E-03	7.83E-02	2.68E-03	3.25E-01
Zr-97		4.54E-03	2.39E-02	2.50E-02	1.02E+00	1.98E-01	
Nb-96		3.50E-07	2.03E-06	2.19E-06	9.00E-05	1.47E-05	
Mo-99		1.40E-03	1.42E-02	1.39E-02	6.13E-01	5.48E-02	
Ru-103+D		5.53E-05	2.00E-03	9.15E-04	5.94E-02	2.13E-03	2.01E-02
Ru-105		1.63E-03	7.05E-03	7.86E-03	2.70E-01	1.05E-01	
Ru-106+D		7.86E-07	3.27E-05	1.36E-05	9.34E-04	3.03E-05	1.05E-01
Sb-129		1.06E-03	4.60E-03	5.15E-03	1.77E-01	6.91E-02	
Te-131		1.51E-03	1.03E-02	1.95E-02	4.45E-01	6.21E-01	
Te-131m		1.76E-04	1.12E-03	1.23E-03	5.07E-02	7.00E-03	
Te-132+D		8.63E-04	9.93E-03	9.18E-03	4.16E-01	3.37E-02	
Te-133m		2.23E-03	1.22E-02	2.04E-02	5.00E-01	5.25E-01	
Te-134		3.32E-03	2.14E-02	3.86E-02	8.98E-01	1.08E+00	
I-131 (ELE)	3.72E+00	1.93E-03	3.29E+00	2.93E-01	1.72E+01	2.23E-01	6.59E-13
I-131 (ORG)							
I-132	1.18E+01	4.26E-03	4.09E+00	6.45E-01	2.48E+01	7.29E-01	
I-133	7.83E+01	4.03E-02	1.43E+01	2.35E+00	1.04E+02	4.61E+00	
I-134	5.84E+02	5.42E-02	1.77E+01	4.29E+00	1.16E+02	1.32E+01	
I-135	2.54E+02	8.48E-02	2.35E+01	3.77E+00	1.51E+02	2.96E+01	
Xe-129m	5.67E-08	8.90E-10	2.02E-08	1.25E-08	3.46E-07	4.30E-09	
Xe-135	6.18E+01	6.52E+00	4.34E+01	4.07E+01	8.95E+02	1.46E+01	
Xe-135m	3.18E+02	6.81E-02	5.22E+00	3.88E+00	5.51E+01	6.79E+00	
Xe-138	7.71E+03	2.84E-01	1.10E+01	2.37E+01	2.47E+02	4.52E+01	
Cs-137+D	2.63E-02	7.88E-06	4.78E-05	4.11E-05	1.10E-03	2.01E-03	1.05E-01
Cs-138	9.78E+03	2.90E+00	1.64E+01	3.01E+01	3.48E+02	3.22E+02	
Ba-139	5.11E+03	1.12E-01	5.06E-01	7.31E-01	1.06E+01	2.22E+02	
Ba-140+D	2.39E+01	5.60E-04	9.96E-03	6.07E-03	3.12E-01	1.16E+00	1.37E-07
Ba-141	1.80E+03	6.48E-04	1.26E-02	2.71E-02	4.52E-01	8.77E+00	
Ba-142	3.53E+02	3.65E-05	3.99E-03	8.25E-03	1.64E-01	3.79E-01	
La-141	3.52E+02	1.05E-02	4.44E-02	4.99E-02	1.66E+00	1.30E+01	
La-142	2.89E+02	8.41E-03	3.85E-02	5.47E-02	1.49E+00	6.01E+00	
Ce-141	1.11E-01	7.63E-05	4.19E-03	1.81E-03	1.24E-01	1.34E-02	1.14E-02
Ce-143		2.64E-03	1.75E-02	1.92E-02	7.97E-01	1.85E-01	
Ce-144+D		1.32E-05	5.46E-04	2.28E-04	1.57E-02	5.61E-04	1.44E+00
Pr-143		3.53E-05	7.19E-03	2.86E-03	2.06E-01	9.22E-04	4.30E-07
Pr-144		1.32E-05	5.46E-04	2.28E-04	1.57E-02	5.53E-04	1.44E+00
U-234		3.68E-07	2.86E-04	3.18E-05	3.02E-05	4.58E-03	9.09E-06
U-235		1.17E-08	9.09E-06	1.01E-06	9.62E-07	1.45E-04	2.90E-07
U-238		1.09E-10	8.46E-08	9.40E-09	8.95E-09	1.35E-06	2.91E-09

Table A-41. (continued)

Radionuclide	FPFRT-9 9/26/58	FPFRT-8 9/18/58	FPFRT-7 9/17/58	FPFRT-6 9/4/58	FPFRT-5 8/27/58	FPFRT-4 8/14/58	FPFRT-3 8/6/58
Ar-41							
Br-84							
Kr-85m							
Kr-87							
Kr-88+D							
Rb-89							
Sr-89	4.02E-04	1.83E+01	1.62E+01	1.94E+01	2.16E+01	6.30E-04	7.34E-04
Sr-90+D	1.88E+00	2.72E-01	2.37E-01	2.37E-01	2.37E-01	1.64E+00	1.72E+00
Sr-91+D							
Sr-92							
Y-91	2.77E-03	2.21E+01	1.95E+01	2.28E+01	2.51E+01	4.01E-03	4.59E-03
Y-92							
Y-93							
Zr-95+D	1.21E-02	3.46E+01	3.30E+01	3.80E+01	4.15E+01	1.57E-02	1.91E-02
Zr-97							
Nb-96							
Mo-99				3.00E-04			
Ru-103+D		3.28E+00	2.31E+00	2.91E+00	3.35E+00		
Ru-105							
Ru-106+D	1.38E-01	1.34E-01	9.36E-02	9.54E-02	9.72E-02	1.44E-01	1.38E-01
Sb-129							
Te-131							
Te-131m							
Te-132+D			8.79E-04	1.42E-02	7.76E-02		
Te-133m							
Te-134							
I-131 (ELE)		1.13E+00	1.24E+00	3.82E+00	7.61E+00		
I-131 (ORG)							
I-132		8.38E-04	1.04E-03	1.68E-02	9.19E-02		
I-133							
I-134							
I-135							
Xe-129m							
Xe-135							
Xe-135m							
Xe-138							
Cs-137+D	4.06E+00	5.39E-01	5.10E-01	5.10E-01	5.10E-01	3.34E+00	3.71E+00
Cs-138							
Ba-139							
Ba-140+D		6.16E+00	5.93E+00	1.21E+01	1.86E+01		
Ba-141							
Ba-142							
La-141							
La-142							
Ce-141		8.49E+00	6.40E+00	8.46E+00	1.00E+01		
Ce-143							
Ce-144+D	2.07E+00	3.17E+00	2.35E+00	2.42E+00	2.48E+00	2.21E+00	2.13E+00
Pr-143		2.08E+00	1.51E+00	2.95E+00	4.45E+00		
Pr-144	2.07E+00	3.17E+00	2.35E+00	2.42E+00	2.48E+00	2.21E+00	2.13E+00
U-234							
U-235							
U-238							

Table A-41. (continued)

Radionuclide	FPFRT-2 8/4/58	FPFRT-1 7/25/58	IET #12 (BOOT) 5/2/58	IET #11 3/20-4/14/58	IET #10(B) 3/1-6/58	IET #10(A) 12/20/57-2/25/58	IET #8 7/31-8/28/57
Ar-41			4.39E+01	3.67E+03	1.31E+03	4.00E+03	1.37E+03
Br-84			4.37E+01	8.57E+00	3.85E+00	1.67E+01	
Kr-85m			2.91E+01	1.50E+01	1.04E+04	8.78E+03	4.08E+00
Kr-87			1.72E+02	3.96E+01	1.97E+04	1.78E+04	1.49E+01
Kr-88+D			1.75E+02	6.56E+01	4.23E+04	3.60E+04	2.07E+01
Rb-89			5.58E+01	4.52E+00	4.17E+02	6.23E+02	2.33E+00
Sr-89	7.13E-04	8.69E-04	4.79E+00	5.64E-03	1.22E+01	6.52E+00	1.30E-03
Sr-90+D	1.62E+00	1.72E+00	2.87E-02	8.60E-06	5.82E-02	2.33E-02	1.51E-06
Sr-91+D			8.35E+01	3.43E-02	8.23E+01	6.98E+01	7.40E-03
Sr-92			1.84E+02	2.02E-02	7.29E+01	6.23E+01	2.09E-03
Y-91	4.47E-03	5.33E-03	4.46E+00	5.35E-04	7.52E+00	2.67E+00	1.71E-06
Y-92			5.88E+01	1.49E-02	8.58E+01	6.98E+01	3.00E-04
Y-93			8.95E+01	1.21E-02	6.93E+01	6.00E+01	1.41E-04
Zr-95+D	1.84E-02	2.17E-02	5.04E+00	6.03E-04	8.06E+00	3.10E+00	
Zr-97			8.50E+01	8.98E-03	5.34E+01	4.72E+01	
Nb-96			8.77E-03	7.93E-07	4.70E-03	4.12E-03	
Mo-99			6.76E+01	5.32E-03	3.62E+01	2.36E+01	2.41E+02
Ru-103+D			3.95E+00	4.61E-04	5.77E+00	2.25E+00	
Ru-105			2.37E+01	2.43E-03	1.32E+01	1.11E+01	
Ru-106+D	1.30E-01	1.40E-01	5.77E-02	7.14E-06	1.06E-01	3.95E-02	
Sb-129			1.55E+01	1.59E-03	8.61E+00	7.23E+00	
Te-131			1.43E+02	4.59E-03	1.23E+01	1.19E+01	
Te-131m			5.38E+00	4.48E-04	2.69E+00	2.28E+00	
Te-132+D			4.45E+01	3.58E-03	2.54E+01	1.54E+01	
Te-133m			1.24E+02	5.04E-03	1.68E+01	1.56E+01	
Te-134			2.49E+02	9.23E-03	2.57E+01	2.47E+01	
I-131 (ELE)			2.19E+01	1.08E+01	2.07E+01	3.42E+01	
I-131 (ORG)							
I-132			5.10E+01	1.72E+01	3.18E+01	6.00E+01	
I-133			1.02E+02	6.85E+01	8.62E+01	3.00E+02	
I-134			4.06E+02	8.67E+01	9.00E+01	2.53E+02	
I-135			1.48E+02	1.00E+02	1.20E+02	4.11E+02	
Xe-129m			4.95E-08	2.16E-08	2.54E-05	1.13E-05	1.01E-09
Xe-135			6.89E+01	6.94E+01	4.72E+04	4.50E+04	6.23E+00
Xe-135m			2.70E+01	1.77E+01	3.18E+02	4.74E+02	1.27E+00
Xe-138			5.61E+01	2.07E+01	1.52E+03	2.51E+03	1.13E+01
Cs-137+D	3.50E+00	3.71E+00	2.95E-02	6.22E-04	9.14E-02	4.86E-02	1.31E-05
Cs-138			4.75E+02	2.86E+01	1.15E+04	1.08E+04	1.03E+01
Ba-139			3.12E+02	9.75E-01	4.69E+02	4.17E+02	2.39E-01
Ba-140+D			2.26E+01	3.87E-03	2.42E+01	1.06E+01	4.01E-04
Ba-141			7.42E+01	2.31E-02	4.97E+00	6.39E+00	4.63E-03
Ba-142			1.06E+01	3.34E-03	4.15E-01	1.08E+00	9.11E-04
La-141			1.48E+02	2.08E-02	8.11E+01	6.80E+01	8.38E-04
La-142			2.63E+02	1.60E-02	6.14E+01	5.44E+01	4.89E-04
Ce-141			8.16E+00	9.61E-04	1.21E+01	4.57E+00	4.68E-07
Ce-143			8.70E+01	7.05E-03	4.24E+01	3.53E+01	
Ce-144+D	2.02E+00	2.20E+00	9.70E-01	1.20E-04	1.76E+00	6.60E-01	
Pr-143			1.27E+01	1.56E-03	1.89E+01	5.91E+00	
Pr-144	2.02E+00	2.20E+00	9.69E-01	1.20E-04	1.76E+00	6.60E-01	
U-234			1.11E-02	4.53E-05	1.23E-04	3.06E-04	
U-235			3.54E-04	1.44E-06	3.91E-06	9.74E-06	
U-238			3.29E-06	1.34E-08	3.64E-08	9.06E-08	

Table A-41. (continued)

Radionuclide	Fuel Element Burn Test B 3/20/57	Fuel Element Burn Test A 3/20/57	IET #6 12/18/56	IET #4(C) 6/29/56	IET #4(B) 5/24-6/29/56	IET #4(A) 5/1-23/56	IET #3 2/11-24/56
Ar-41			8.05E+03		6.06E+03	1.81E+03	2.25E+03
Br-84			1.17E-02	1.32E+03	3.83E+03	7.17E+01	7.50E+02
Kr-85m			6.54E+00	9.94E+02	6.94E+03	2.06E+02	8.49E+02
Kr-87			2.13E+00	3.84E+03	3.05E+04	7.74E+02	4.28E+03
Kr-88+D			1.66E+01	4.98E+03	3.74E+04	1.07E+03	4.67E+03
Rb-89			2.40E-05	4.78E+03	3.60E+03	3.87E+01	1.63E+03
Sr-89	4.56E+00	5.83E-01	6.93E+00	4.50E+02	3.35E+00	1.21E-01	3.37E+01
Sr-90+D	1.27E+00	9.25E-03	4.80E-02	3.71E+00	3.99E-03	1.26E-04	2.28E-01
Sr-91+D			6.50E+01	4.30E+03	3.04E+01	9.30E-01	3.69E+02
Sr-92			2.56E+01	8.33E+03	3.61E+01	1.03E+00	7.90E+02
Y-91	8.20E+00	7.10E-01	7.05E+00	4.70E+02	1.90E-02	7.54E-04	3.38E+01
Y-92			6.99E+01	5.32E+03	1.66E+01	5.66E-01	3.48E+02
Y-93			7.15E+01	4.46E+03	1.48E+01	4.56E-01	3.84E+02
Zr-95+D	1.09E+01	7.63E-01	7.59E+00	5.11E+02	1.15E-01	3.63E-03	3.66E+01
Zr-97			6.58E+01	2.88E+03	8.64E+00	2.68E-01	2.83E+02
Nb-96			6.10E-03	2.32E-01	6.41E-04	2.00E-05	2.51E-02
Mo99		4.35E-07	4.24E+01	1.43E+03	2.39E+00	7.52E-02	1.79E+02
Ru103+D	1.30E+00	3.70E-01	5.58E+00	3.48E+02	9.14E-02	2.88E-03	2.72E+01
Ru-105			8.08E+00	1.23E+03	4.40E+00	1.31E-01	1.06E+02
Ru-106+D	1.53E+00	1.64E-02	9.49E-02	7.16E+00	1.30E-03	4.09E-05	4.52E-01
Sb-129			5.23E+00	8.11E+02	2.89E+00	8.55E-02	7.04E+01
Te-131			8.18E-01	4.09E+03	1.61E+01	3.34E-01	5.06E+02
Te-131m			3.52E+00	1.22E+02	3.09E-01	9.78E-03	1.40E+01
Te-132+D		4.51E-06	2.88E+01	9.89E+02	1.47E+00	4.62E-02	1.25E+02
Te-133m			7.55E-01	4.20E+03	1.63E+01	3.81E-01	4.98E+02
Te-134			4.13E-01	8.62E+03	3.12E+01	6.62E-01	1.05E+03
I-131 (ELE)	3.06E-06	2.96E-01	1.35E+01	5.47E+02	2.57E+02	8.11E+00	3.21E+02
I-131 (ORG)							
I-132		6.86E-05	2.93E+01	8.57E+02	6.09E+02	1.70E+01	5.19E+02
I-133			7.23E+01	2.65E+03	6.56E+03	2.04E+02	1.33E+03
I-134			4.94E+00	1.18E+04	3.44E+04	7.91E+02	5.55E+03
I-135			6.18E+01	5.85E+03	1.98E+04	6.00E+02	2.45E+03
Xe-129m	9.42E-14	1.91E-08	2.66E-08	1.13E-06	1.48E-06	4.68E-08	1.30E-06
Xe-135			9.17E+01	1.96E+03	8.26E+03	2.72E+02	1.78E+03
Xe-135m			1.06E+01	1.11E+03	4.33E+03	1.13E+02	7.42E+02
Xe-138			5.99E-06	4.65E+03	1.57E+04	1.53E+02	5.79E+03
Cs-137+D	1.29E+00	9.38E-03	4.86E-02	3.76E+00	3.08E-02	9.71E-04	2.34E-01
Cs-138			1.34E-01	1.03E+04	2.68E+04	6.16E+02	3.89E+03
Ba-139			7.85E+00	1.06E+04	6.41E+02	1.75E+01	1.22E+03
Ba-140+D	3.81E-04	1.60E-01	2.40E+01	1.13E+03	1.44E+00	4.54E-02	1.19E+02
Ba-141			2.58E-04	5.45E+03	1.97E+01	2.50E-01	6.82E+02
Ba-142			2.57E-08	3.25E+03	4.72E+00	3.14E-02	4.05E+02
La-141			4.34E+01	7.41E+03	2.94E+01	8.92E-01	6.38E+02
La-142			1.02E+01	9.57E+03	4.05E+01	1.09E+00	1.02E+03
Ce-141	1.06E+00	6.45E-01	1.20E+01	6.94E+02	6.58E-02	2.40E-03	5.63E+01
Ce-143		1.20E-14	5.54E+01	1.89E+03	4.64E+00	1.46E-01	2.22E+02
Ce-144+D	2.20E+01	2.65E-01	1.59E+00	1.19E+02	2.18E-02	6.88E-04	7.57E+00
Pr-143	8.61E-04	1.98E-01	1.82E+01	9.52E+02	2.40E-02	8.82E-04	9.56E+01
Pr-144	2.20E+01	2.65E-01	1.59E+00	1.19E+02	2.16E-02	6.86E-04	7.55E+00
U-234	1.34E-04	1.34E-05	3.18E-04	3.30E-02			1.14E-02
U-235	4.32E-06	4.32E-07	1.01E-05	1.05E-03			3.62E-04
U-238	6.75E-08	6.75E-09	9.40E-08	9.78E-06			3.36E-06

Table A-41. (continued)

Radionuclide	NRF S1W Engineering Test 6/18-7/1/55	BORAX-I Excursion 7/22/54
Ar-41		
Br-84	2.38E-02	1.46E-01
Kr-85m	2.48E+01	2.38E-01
Kr-87	1.13E+01	1.22E+01
Kr-88+D	4.86E+01	2.18E+01
Rb-89	6.02E-06	8.24E+01
Sr-89	7.38E-03	1.45E-01
Sr-90+D	4.50E-05	2.17E-03
Sr-91+D	1.37E-01	2.13E+01
Sr-92	7.38E-02	2.42E+01
Y-91	7.89E-03	1.46E-02
Y-92	9.21E-02	4.38E+00
Y-93	1.48E-01	2.52E+00
Zr-95+D	7.99E-03	1.22E-02
Zr-97	1.37E-01	7.42E-02
Nb-96	1.17E-05	5.39E-06
Mo-99	9.62E-02	1.96E-02
Ru-103+D	1.20E-01	3.05E-03
Ru-105	3.48E-01	4.53E-02
Ru-106+D	1.80E-03	1.20E-03
Sb-129	1.33E-02	3.00E-02
Te-131	3.79E-03	4.48E-01
Te-131m	1.74E-01	2.37E-03
Te-132+D	1.26E+00	1.21E-02
Te-133m	1.60E-01	3.54E-01
Te-134	1.15E-01	8.29E-01
I-131 (ELE)	4.74E+00	5.55E-03
I-131 (ORG)		
I-132	4.92E+00	9.46E-03
I-133	2.03E+01	5.60E-02
I-134	1.40E+00	8.77E-01
I-135	1.49E+01	4.65E-01
Xe-129m		2.46E-09
Xe-135	1.73E+02	1.76E+00
Xe-135m	1.56E-03	3.89E+00
Xe-138	3.86E-03	9.21E+01
Cs-137+D	9.00E-04	1.69E-03
Cs-138	3.54E-02	2.47E+02
Ba-139	3.27E-02	1.47E+02
Ba-140+D	3.47E-02	7.59E-01
Ba-141	3.88E-05	2.82E+01
Ba-142	1.30E-07	3.47E+00
La-141	9.37E-02	1.29E+01
La-142	3.79E-02	9.32E+00
Ce-141	1.40E-02	1.03E-02
Ce-143	1.30E-01	3.62E-02
Ce-144+D	1.50E-03	1.75E-02
Pr-143	3.08E-02	3.04E-03
Pr-144	2.60E-05	1.74E-02
U-234		2.81E-04
U-235		9.09E-06
U-238		1.42E-07

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B.1 INTRODUCTION

The second essential element needed to perform a dose evaluation, after the source term, is the estimate of meteorological dispersion. Because the dose to an individual depends on the ground-level concentration of radionuclides where the individual is located, an estimate must be made of the diffusion of the radionuclides between the release point and the receptor. In the late 1960s and early 1970s pioneering research was conducted at the Idaho National Engineering Laboratory (INEL) [then the National Reactor Testing Station (NRTS)] to develop models to more accurately predict dispersion. The presently used model, MESODIF, is an outgrowth of that early research and is described below. The MESODIF model was used at the INEL to estimate offsite doses from 1973 to the present. It continues to be used to predict the consequences of accidental releases at the INEL, for both actual events and for exercises. One key to the ability of the model to correctly predict INEL dispersion is a good network of towers that provide the input of necessary meteorological parameters. The INEL is recognized as having one of the most extensive meteorological data bases existing for a regional setting the size of the Upper Snake River Plain.

B.2 MESODIF MODEL

The MESODIF model is an objective regional trajectory computational scheme that has been combined with the Gaussian diffusion equation for a continuous point source. The trajectory computations are from a simple interpolation method that is driven empirically by data from a network of 20 or more tower-mounted wind sensors. Trajectory validating research has been conducted by NOAA, Idaho Falls, and has been reported in various research reports (Wendell, 1970; Wendell, 1972). Numerous atmospheric diffusion research studies have been conducted by NOAA over the INEL and Upper Snake River Plain; these results are well documented and have been published in numerous journals and manuscripts (Yanskey et al., 1966; Islitzer and Dumbauld, 1963; Islitzer, 1961; Lewellen et al., 1985; Start et al., 1985; Sagendorf and Fairobent, 1986; Hoff et al., 1987; Bowman, 1984; Clawson et al., 1989). The MESODIF model is a forward time-marching Gaussian plume model in which successive, small plume elements are advected throughout the computational area (the Upper Snake River Plain). The MESODIF model is one of three types of atmospheric transport and diffusion models identified as acceptable to the U. S. Nuclear Regulatory Commission (NRC) staff for use in providing assessments of potential exposures because of airborne effluents (NRC, 1977). Hourly values of stability, mixing depth, and winds from the array of towers, along with the source description, are the model inputs. The output information consists of time integrated concentrations determined for a computation mesh of about 200 points. The model is driven by the regional time and space varying meteorological parameters instead of using a joint-frequency stability wind rose. Therefore, MESODIF, which uses varying meteorological data fields, avoids some of the conceptual biases of joint-frequency based models (Start and Wendell, 1974). Some of the obvious biases can result from (a) changes of stability category during travel to a receptor, which is not a reversible or self-compensating phenomena; (b) flow stagnation; (c) flowfield reversals; and (d) other spatial and temporal meteorological variations. MESODIF handles these effects through the empirical time history of the input data; the joint frequency distribution models cannot account for these phenomena.

B.3 METHODS

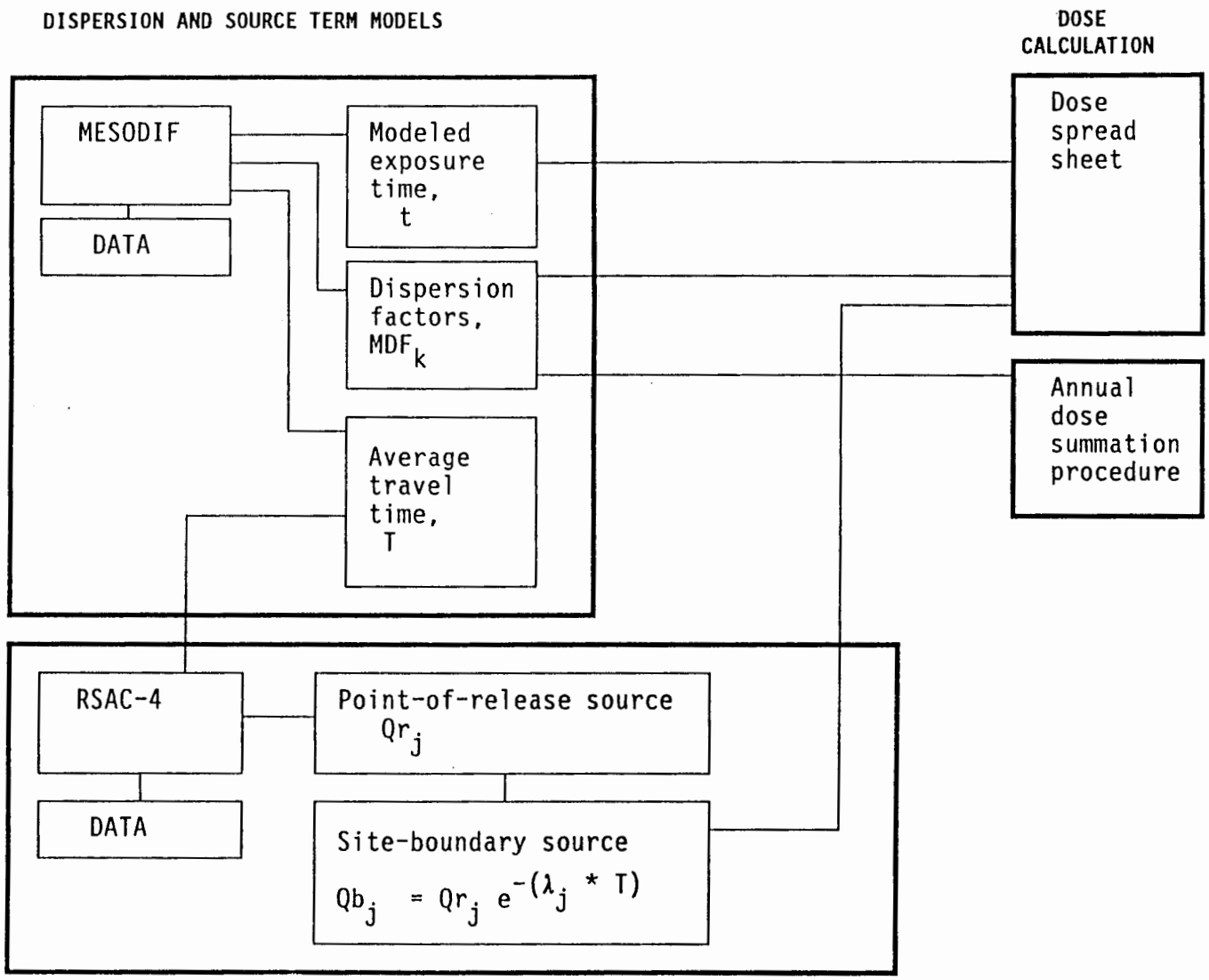
Since 1973, meteorological information gathered from the 26-station INEL telemetry network in the Upper Snake River Plain has been evaluated annually using the MESODIF computer code. The results were illustrated in dispersion isopleth maps contained in annual INEL Site environmental monitoring reports. Adequate meteorological information was not available before 1973 or in 1978, when the telemetry system was being upgraded, to produce annual dispersion isopleth maps.

For these years, the MESODIF computer code was run on a 9-year (1974-1983 but excluding 1978) set of meteorological data from the telemetry system. The resulting "average dispersion conditions" were used in computing the annual impact from operational releases before 1973 and in 1978. The annual dispersion isopleth maps prepared from the MESODIF computer code output were used from 1973 through 1989, except for 1978.

The annual average wind speed toward the inhabited location of highest annual average concentration from a point midway between the Idaho Chemical Processing Plant (ICPP) and Test Reactor Area (TRA) was estimated based on the joint frequency distribution of wind speeds. The wind speed distribution for 9 years (1974-1983) was used to estimate the annual average wind speed before 1984. The annual distributions were used for this purpose beginning in 1984.

The detailed data from a grid of 176 points, on which the dispersion isopleth map is based, were used to calculate the ground-level dispersion coefficient at Atomic City, the location of the maximally exposed individual for the 9-year average. This was done with a log-log interpolation from the four grid points surrounding Atomic City.

MESODIF was also used to calculate dispersion coefficients for episodic events occurring at the INEL. The data flow for discrete episodic events is shown in Figure B-1. For these individual events MESODIF was run on a finer scale grid, grid point spacing of 2 mi as opposed to 5 1/3 mi for the annual operations. Available meteorological data for the years before 1967 were limited to wind sensor tower sites located at the Central Facilities Area



(FOR: 52 radionuclides where j = 1,52 ; 16 locations where k = 1,16)

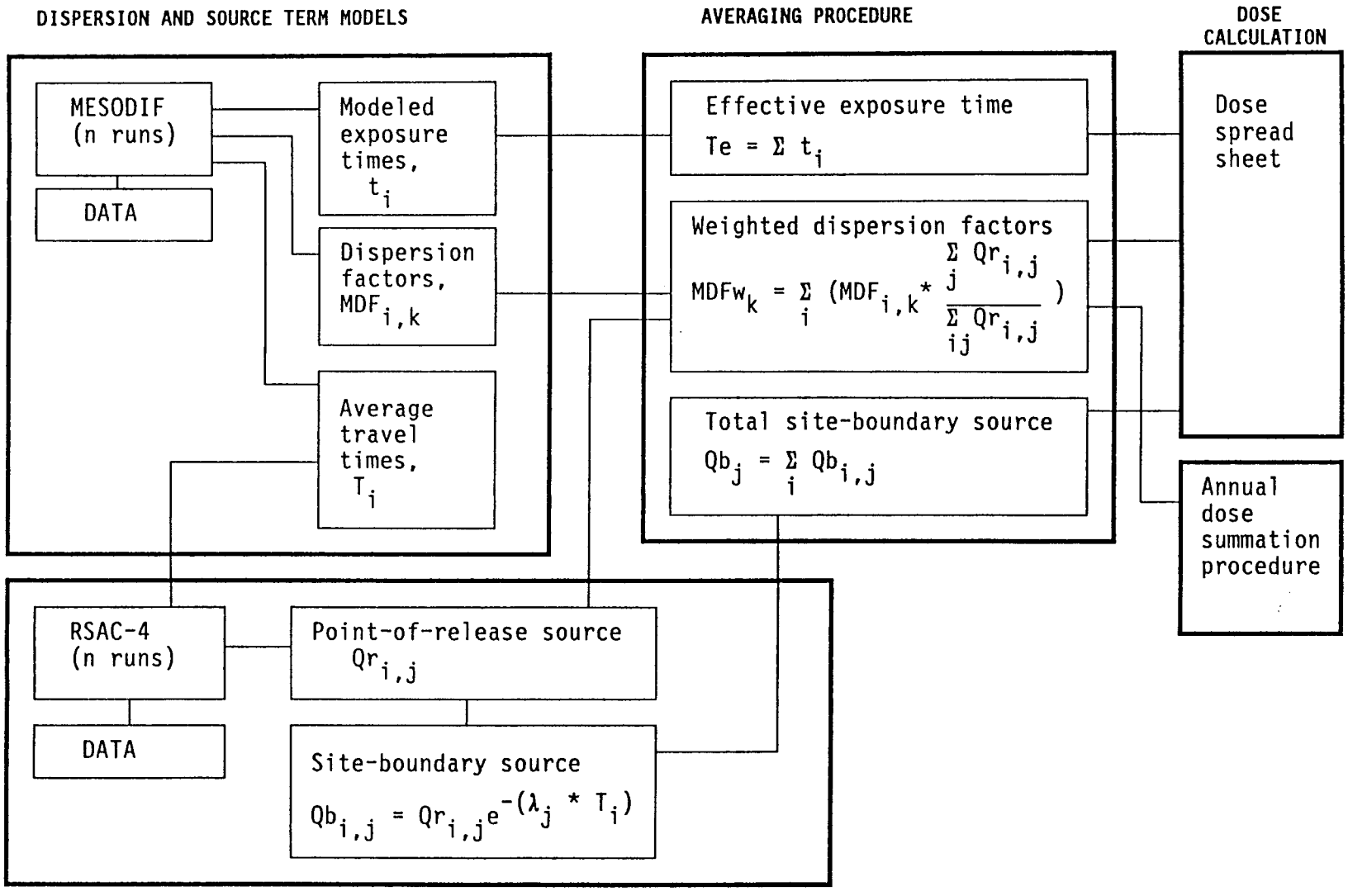
Figure B-1. Data flow for discrete episodic releases.

(CFA) and Initial Engine Test (IET). The accuracy of the plume trajectories calculated for the past events depends on how well these two wind stations were able to represent the actual winds affecting the plume. It is reasonable to assume that the further the plume traveled from the wind tower locations, the less reliable were the calculations. Most of the events were initiated within a few miles of one of these sites, so the important initial portions of these calculated trajectories are considered reasonable representations. Because the primary interest of this project was the radioactive dose received by an inhabitant near the INEL Site boundaries, the area of concern is limited to areas relatively close to the available meteorological data sites.

The elapsed time from the release of the first puff until the first puff and all subsequently released puffs had left the calculation grid was reported as the modeled time. The transit time to the INEL Site boundary for episodic releases was estimated to be the average of the transit times for the successive plume elements (or puffs) to reach the Site boundary.

Calculating dispersion coefficients for episodic releases associated with some of the Aircraft Nuclear Propulsion Program (ANP) tests presented a difficult technical problem. Several of the Initial Engine Tests (IETs) were performed over extended periods of time ranging from a few weeks to several months. Depending on the test protocols, the reactor was operated in a noncontinuous manner at differing times during the day and for varying lengths of time. In some instances, the relative composition of radionuclides in the effluent also changed as the test progressed. Two methods were applied to estimate the dispersion coefficient for the extended tests. If detailed information concerning reactor operations, release rates, and release times was not available, the test was divided into one or more operational periods, and an average dispersion coefficient was calculated over the entire length of each period. This method was used for IETs #4, #6, and #16.

When detailed information concerning reactor operations and radioactive releases for an IET was available, dispersion factors were calculated for each of the individual runs within a test, as shown in Figure B-2. A weighted-average dispersion coefficient was then calculated for the test. The



(FOR: n releases where i = 1,n ; 52 radionuclides where j = 1,52 ; 16 locations where k = 1,16)

Figure B-2. Data flow for prolonged episodic releases.

weighted-average dispersion coefficient was equal to the sum of products of the dispersion coefficient for each run multiplied by the total radioactivity released during the run (expressed as 10-minute-decayed curies) divided by the total radioactivity released during the test. If the relative composition of radionuclides in the effluent changed during a test, then the test was divided into multiple periods, and weighted-average dispersion coefficients were calculated for each period of operation. Weighted-average dispersion coefficients were calculated for IETs #3, #8, #10, #11, #14, #15, #17 through #21 (FEET #1), #23 (FEET #2), #25, and #26. IETs #12 (BOOT), #13 (HTRE No. 3 Excursion), #22 (LIME), and #24 (LIME-II) were short duration releases and were modeled as such.

To obtain a dispersion coefficient for a reasonable number of points on the INEL Site boundary and nearby communities, 16 points were chosen. Most of these points represent locations where there is reason to believe someone lived in the early days of INEL operations, as noted by buildings on a 1961 map of the NRTS vicinity. The locations and titles are shown in Figure B-3.

There are three locations that were chosen to fill in around the Site perimeter and probably do not represent occupied residences: (1) Cerro Grande, an abandoned railroad stop; (2) Frenchman's Cabin, an abandoned cabin near Big Southern Butte; and (3) Cellar, a potato cellar southeast of the INEL Site. In almost all cases, after a pollutant plume has traveled 10 mi, concentrations decrease with distance from the source of the pollutant. Thus, including additional locations at larger distances from the INEL Site boundary was not warranted. For short-term releases, dispersion conditions and isopleth maps were reviewed to ensure that the plume had not passed between the chosen locations on the Site boundary, resulting in unrealistically low estimates of the dispersion coefficient at occupied residences.

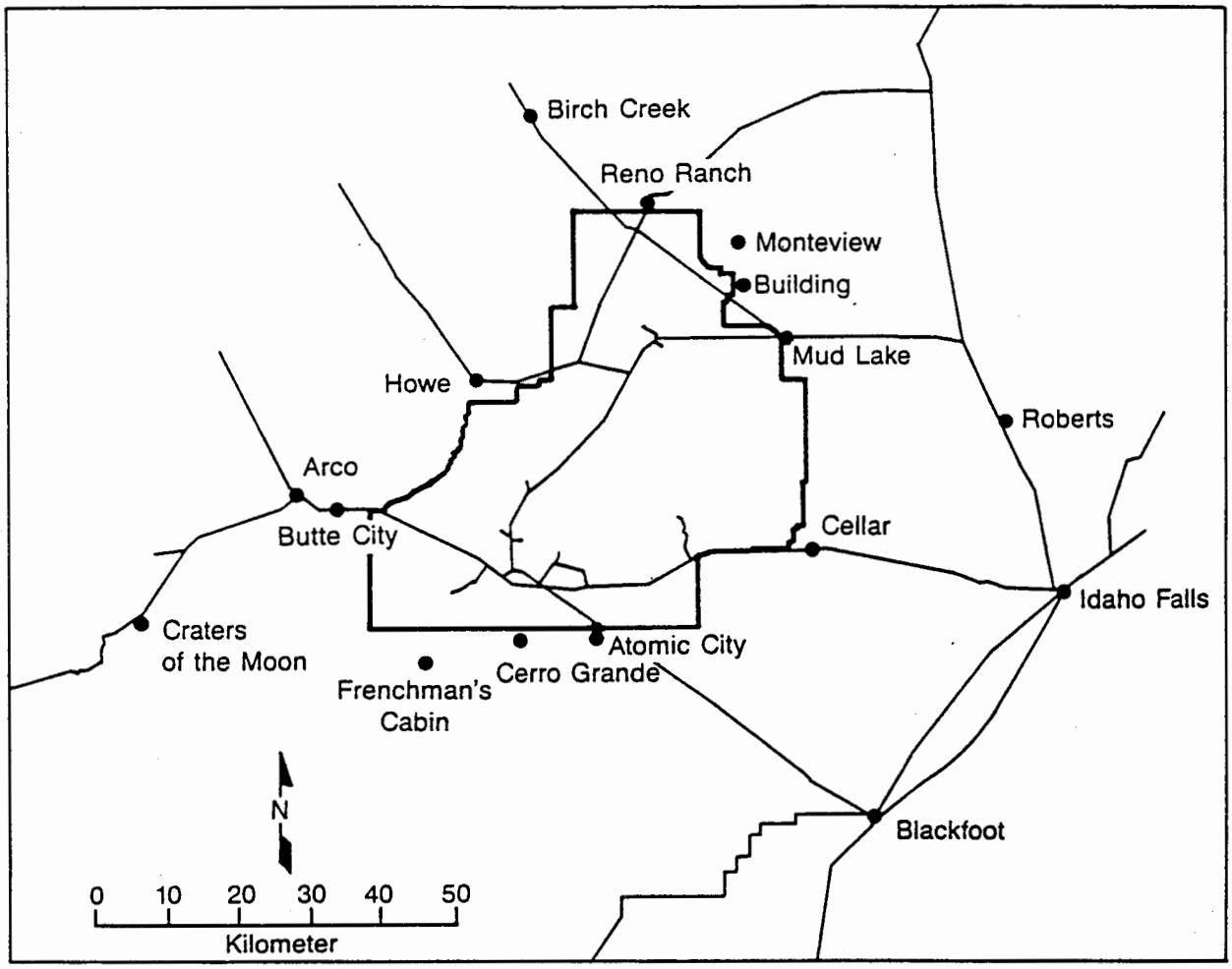


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Figure B-3. Locations for dispersion calculations for episodic releases.

B.4 RESULTS OF DISPERSION CALCULATIONS

The normalized ground-level concentrations for the 9-year average (1974-1983) are shown in Figure B-4. To facilitate the display, the dispersion coefficient values are given in whole numbers and must be multiplied by 10^{-9} h^2/m^3. To obtain the average air concentration (Ci/m^3 or $\mu Ci/mL$) for a radionuclide released from the center of the INEL along any dispersion coefficient isopleth in Figure B-2, multiply that value (in units of h^2/m^3) by the average release rate (in units of Ci/h) of the radionuclide released during the year and divide by the number of hours in a year (8760 hours).

The dispersion isopleths from 1989 through 1973 are shown in Figures B-5 through B-20, reproduced from references cited in the figure captions. The maximum ground-level air concentration and the dispersion pattern remain fairly constant from year to year. Thus, the use of the average dispersion coefficient was appropriate for early years when an extensive set of meteorological data was not available. The dispersion coefficient and the location of the maximally exposed hypothetical offsite individual are summarized in Table B-1 for each year of INEL Site operation between 1952 and 1989. The locations of Frenchman's Cabin and Cerro Grande are unoccupied for most or all of the year and are not assumed to be the location of the maximally exposed offsite individual for operational releases.

The episodic releases were individually modeled to find the location of the maximally exposed offsite individual and the dispersion coefficient. These are summarized in Table B-2. A trajectory map, dispersion map, and table are shown for the more important releases (Figures B-21 through B-34 and Tables B-3 through B-9). IET #4(B) (Figure B-32 and Table B-8) is included because it represents one of the longer episodic releases modeled, even though the calculated dose was lower than the other tests shown.

The dispersion map for the SL-1 Accident differs from the other maps in that it is based on extensive environmental sampling at the time of the accident and not a modeled dispersion estimate. This is the only episodic release where the Task Group believed the environmental data were sufficiently

complete to warrant its use, rather than a dispersion factor calculated from the meteorological data. Because the original map was not reproducible, the map was redrafted, as accurately as possible, from Horan and Gammill (1961).

The dispersion coefficient for the episodic releases differs somewhat from the annual dispersion coefficient. The units are in h/m^3. The diffusion coefficient is the time-integrated ground-level concentration from a unit release. In this case, the dispersion coefficient must be multiplied by the number of curies released and divided by the hours for which the calculation was performed, shown as "hours modeled" in Table B-2. Each calculation was continued until the material reached the edge of the calculational grid. This means that the number of hours of the calculation varies from 0.9 to 874 hours. All short-term dispersion coefficients were normalized for a 1 Ci release even if the release occurred over an extended time period. In most cases calculated here, the radioactive plume makes one pass over the receptor location, and the immersion and inhalation radioactive doses are realized in those minutes of plume passage. The model also accounts for stagnation over the receptor or multiple passes. The number of hours of the calculation is not important, but the exposure period of the individual to the "average" air concentration must be the same as the number of hours in the calculation.

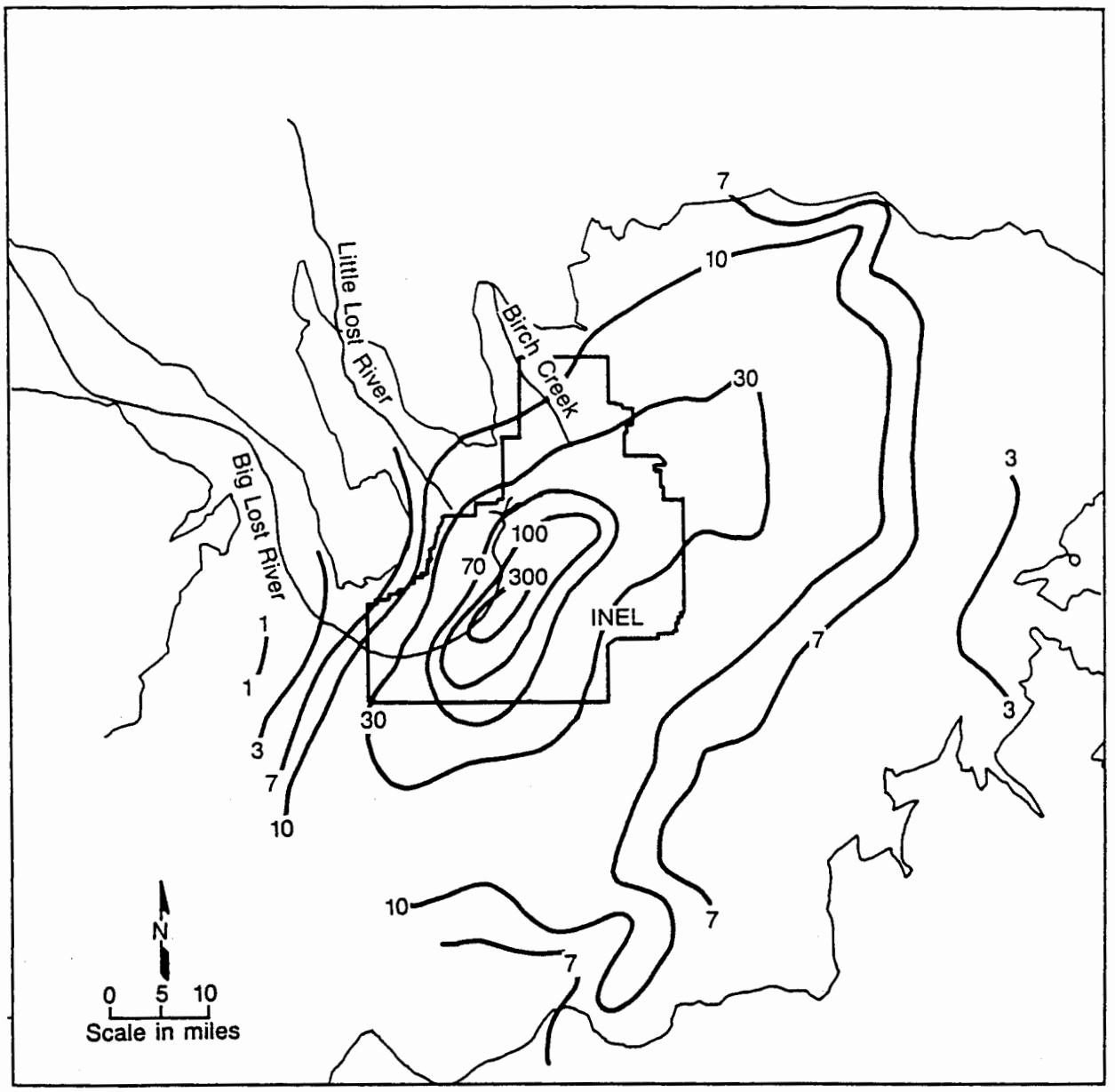

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Figure B-4. Nine-year (1974-1983) average mesoscale dispersion isopleths of air concentration at ground level, normalized to unit release rate (Bowman, 1984).

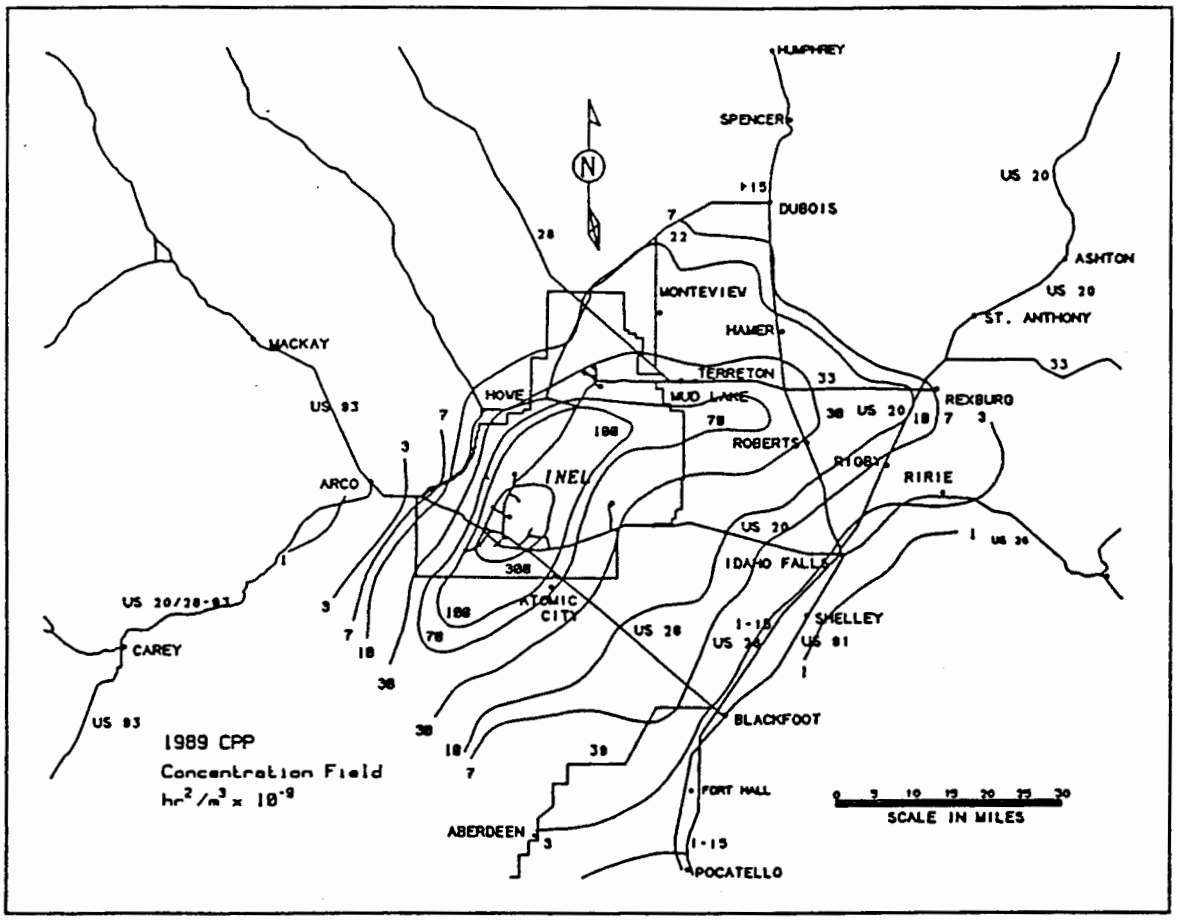

Figure B-5. 1989 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Hoff et. al, 1990).

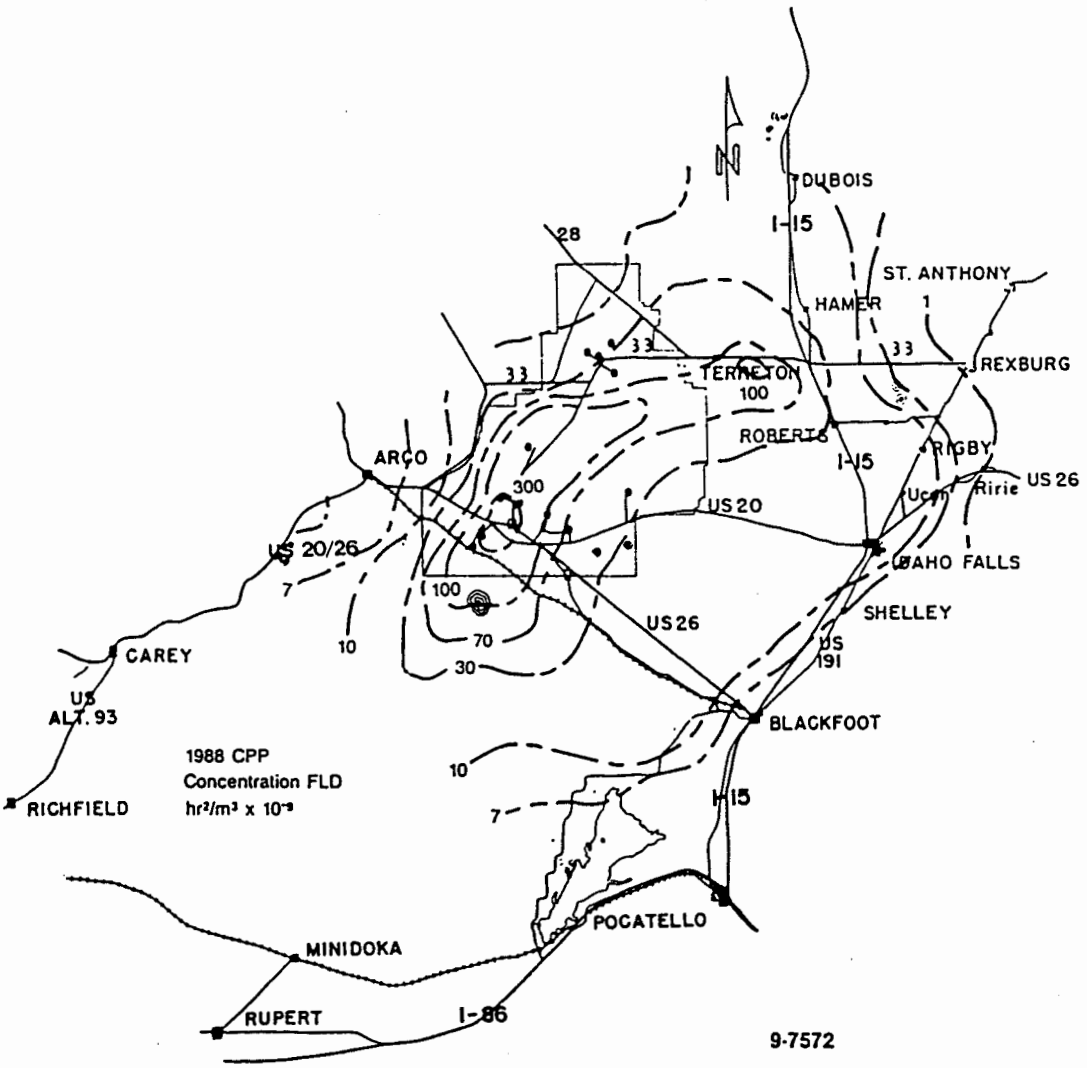

Figure B-6. 1988 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Hoff et. al, 1989).

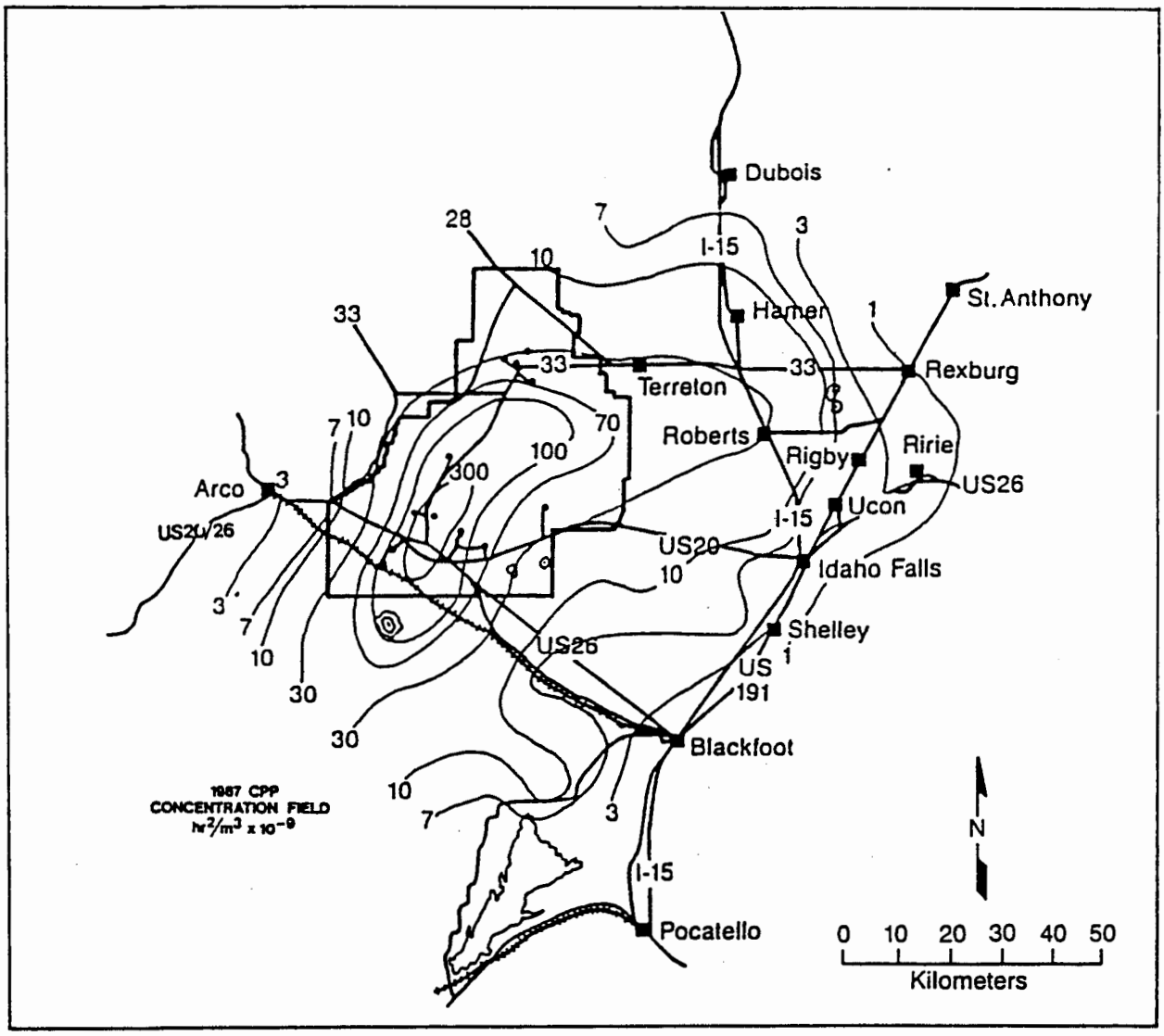

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Figure B-7. 1987 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Chew and Mitchell, 1988).

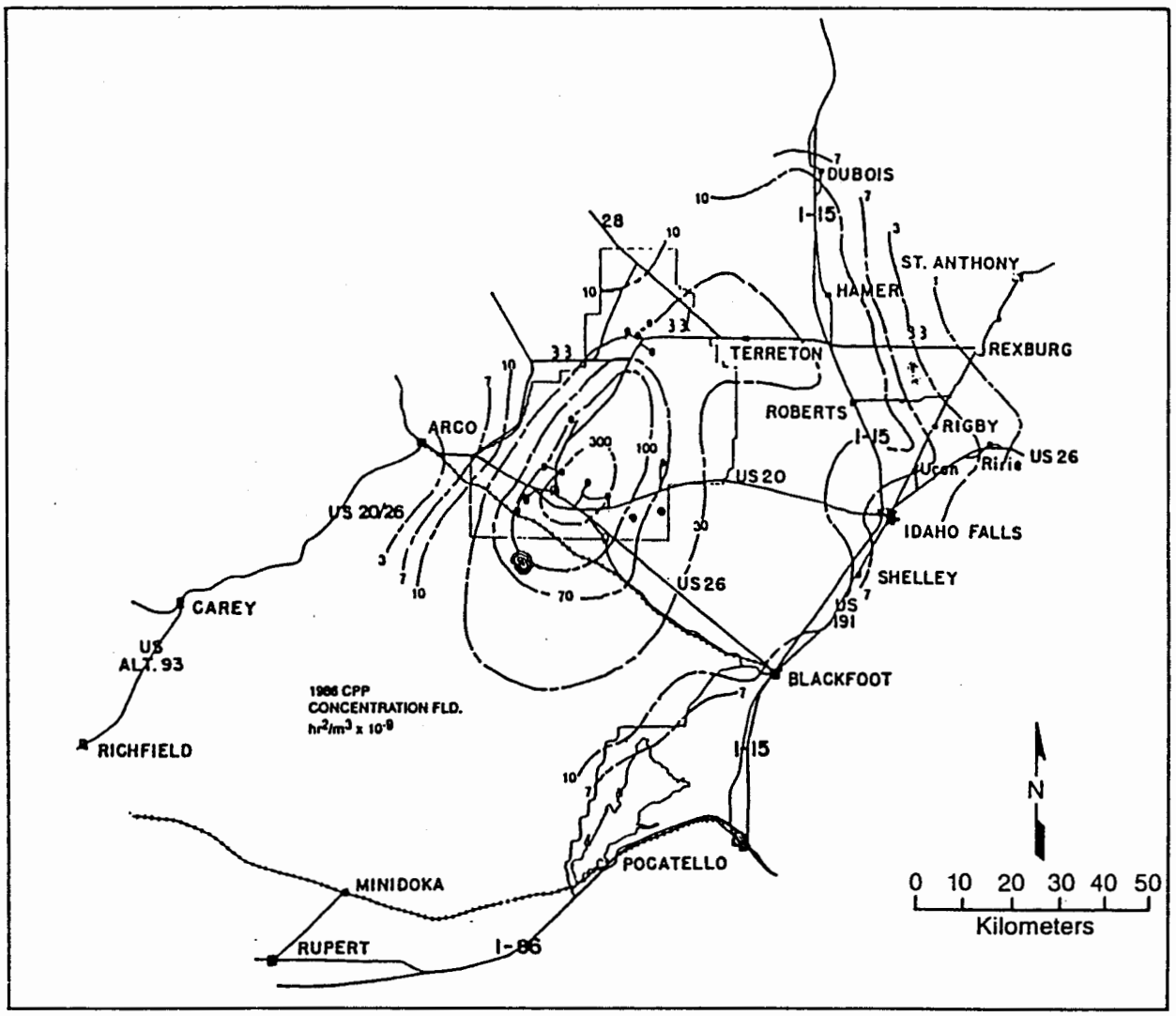

Figure B-8. 1986 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Hoff et al., 1987).

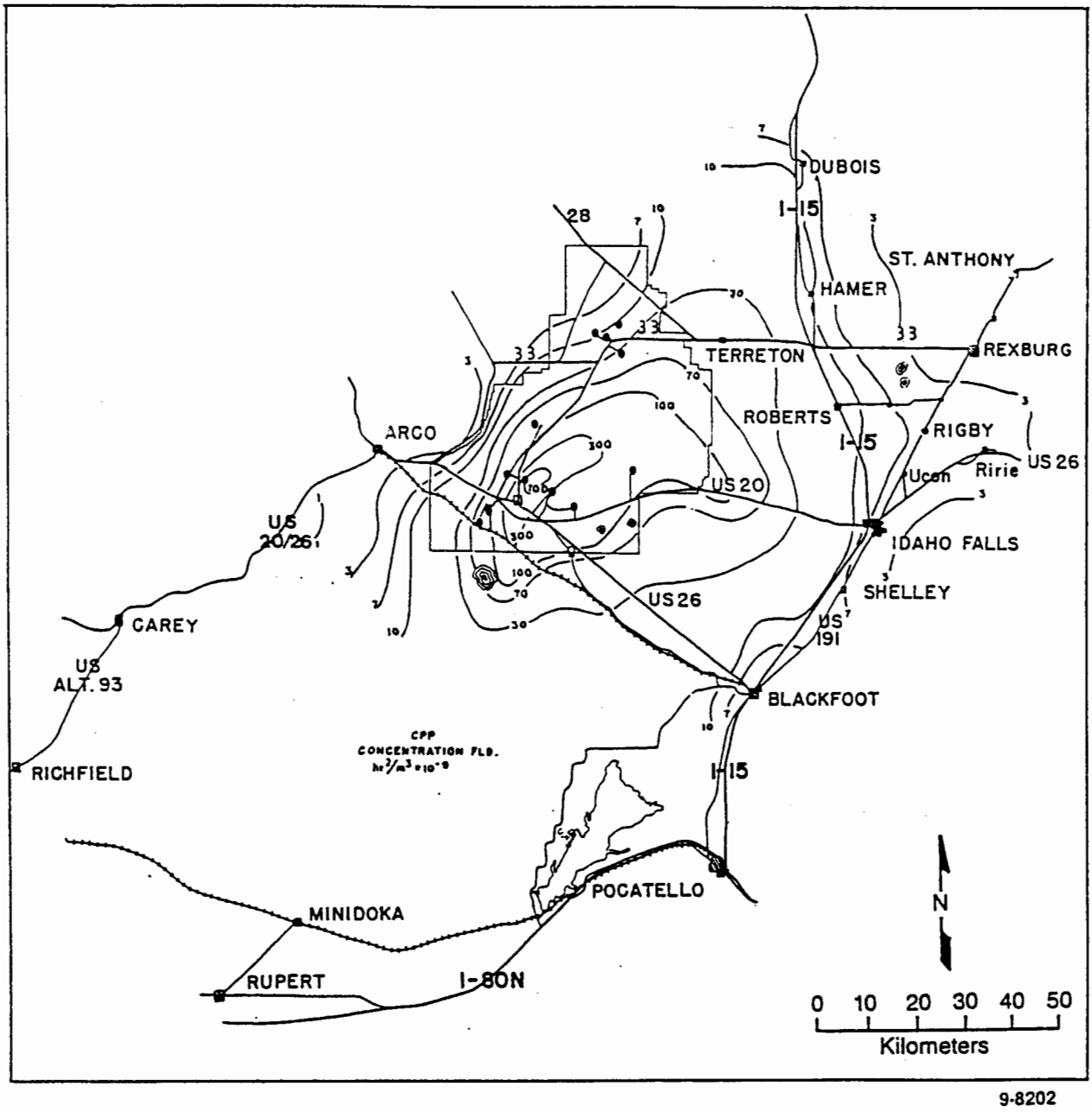

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Figure B-9. 1985 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Hoff et al., 1986).

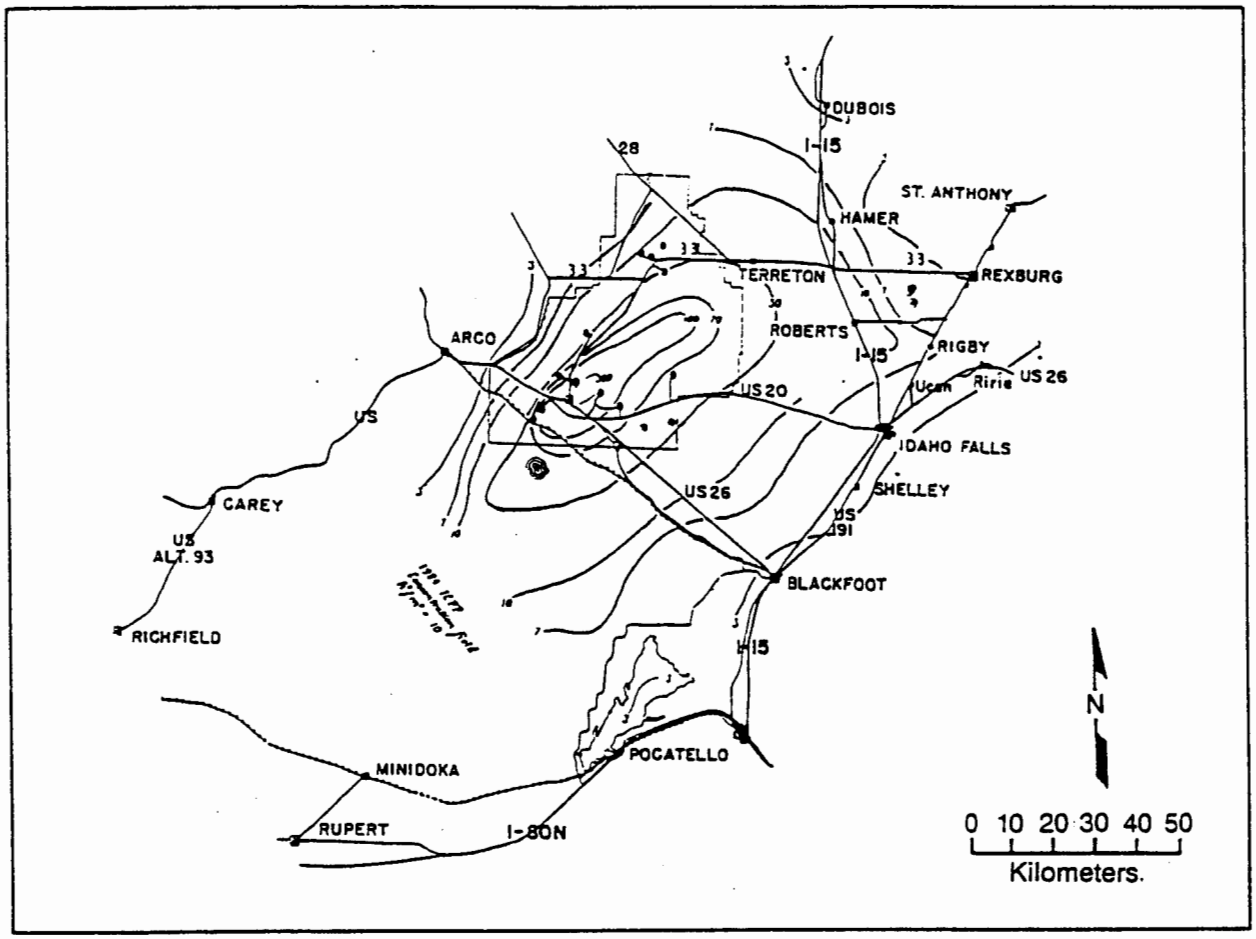

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Figure B-10. 1984 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Hoff et al., 1985).

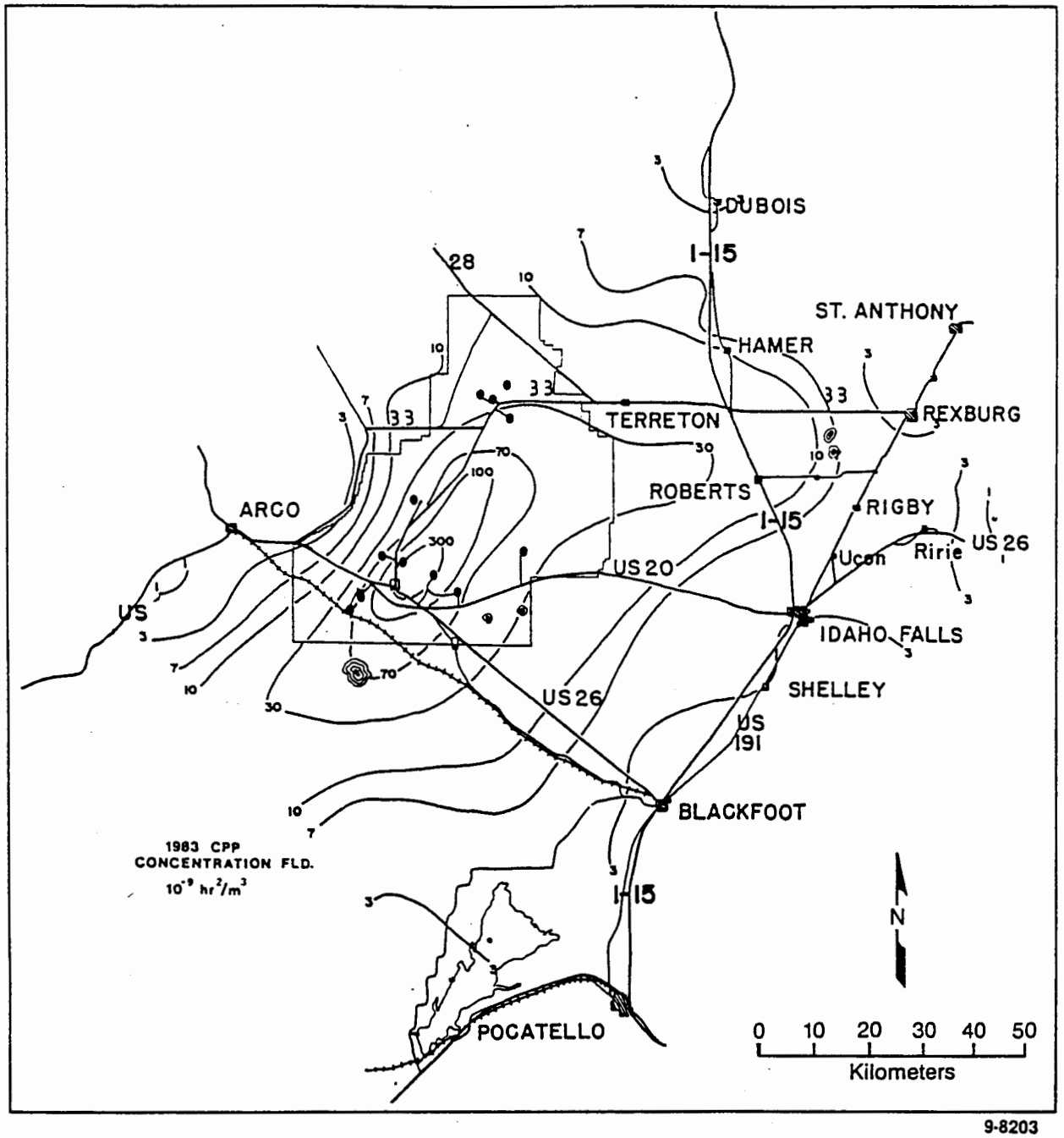

Figure B-11. 1983 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (Hoff et al., 1984).

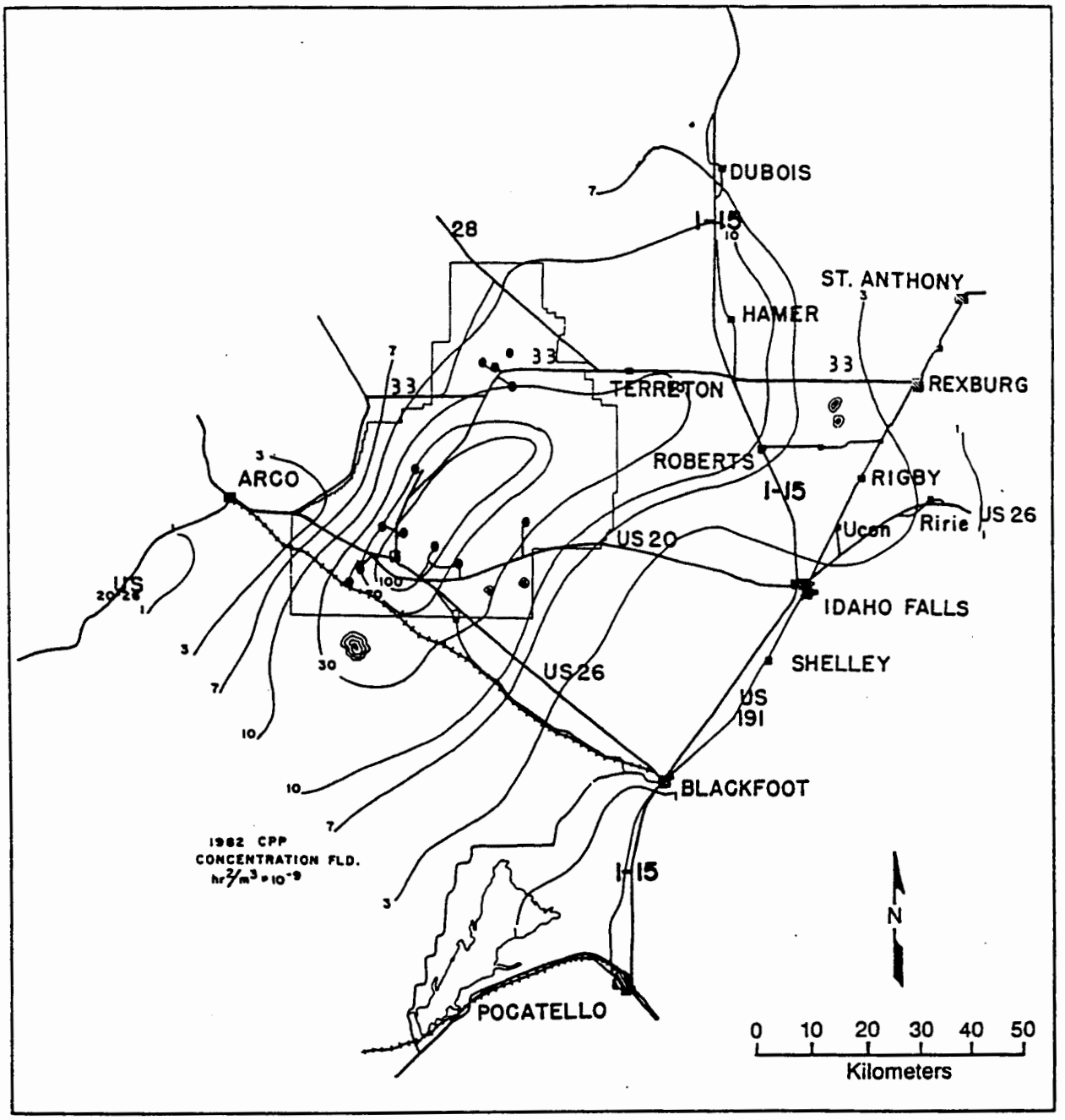

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Figure B-12. 1982 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (DOE, 1983).

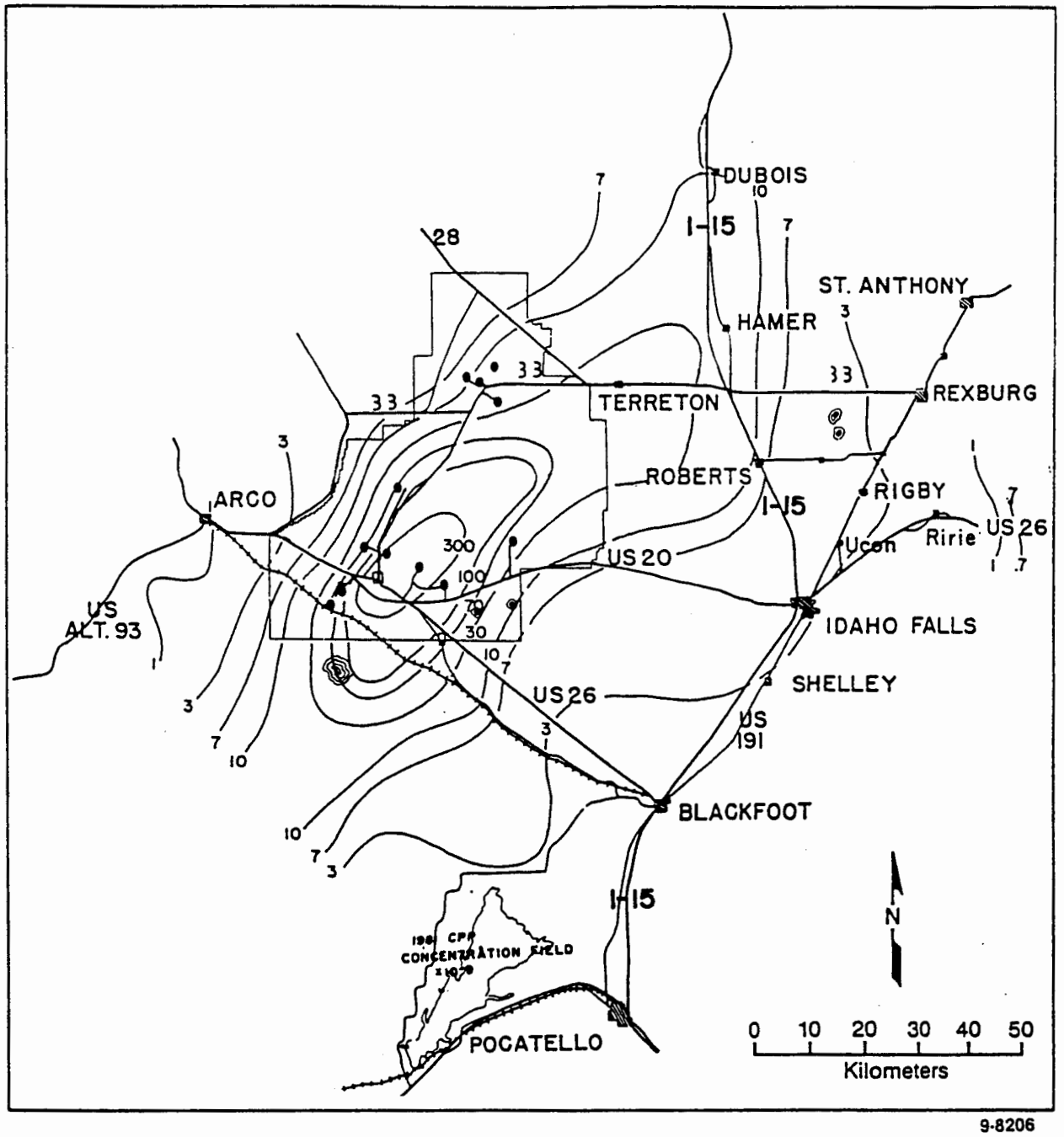

Figure B-13. 1981 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (DOE, 1982).

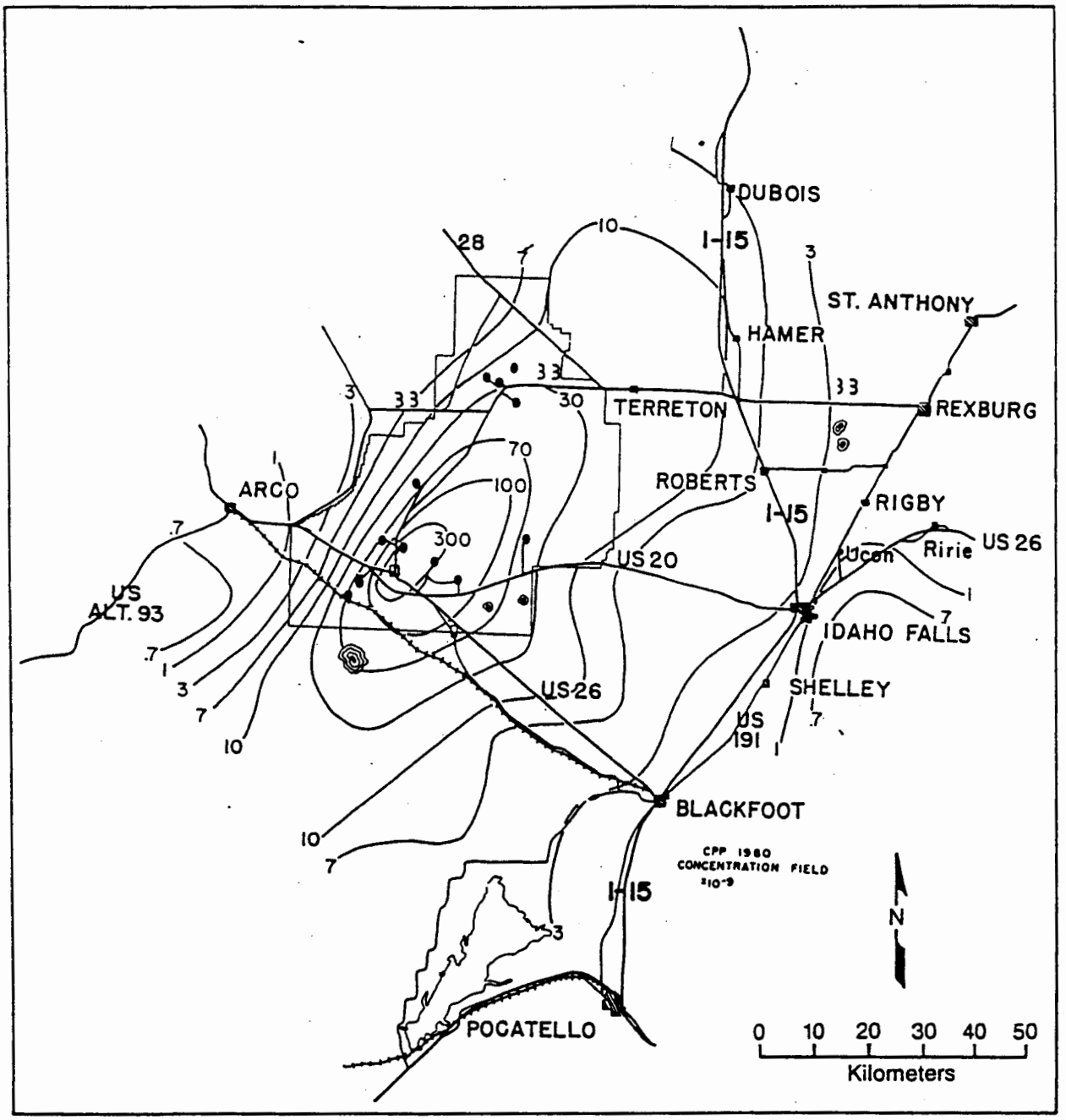

Figure B-14. 1980 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (DOE, 1981).

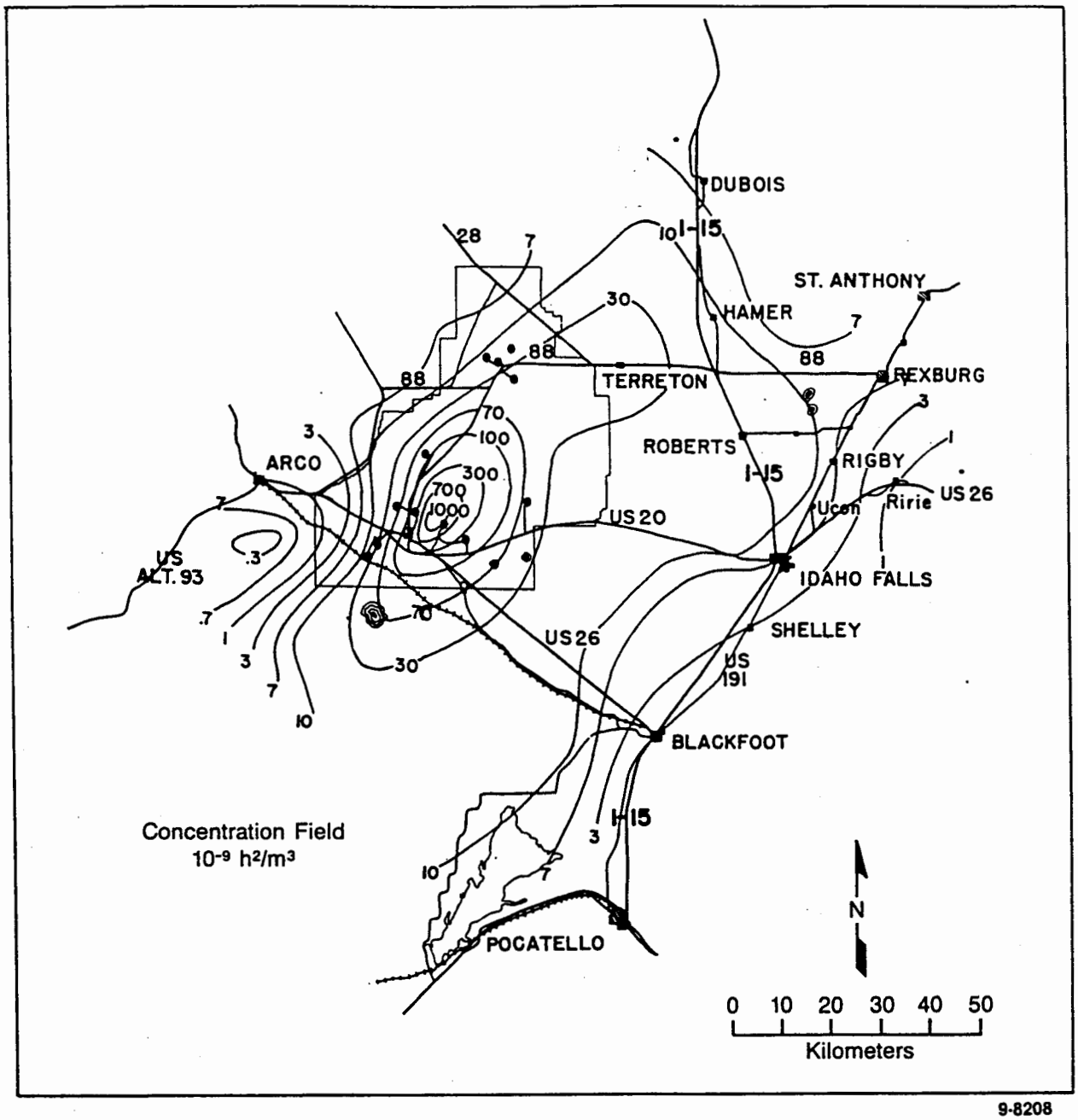

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Figure B-15. 1979 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (DOE, 1980).

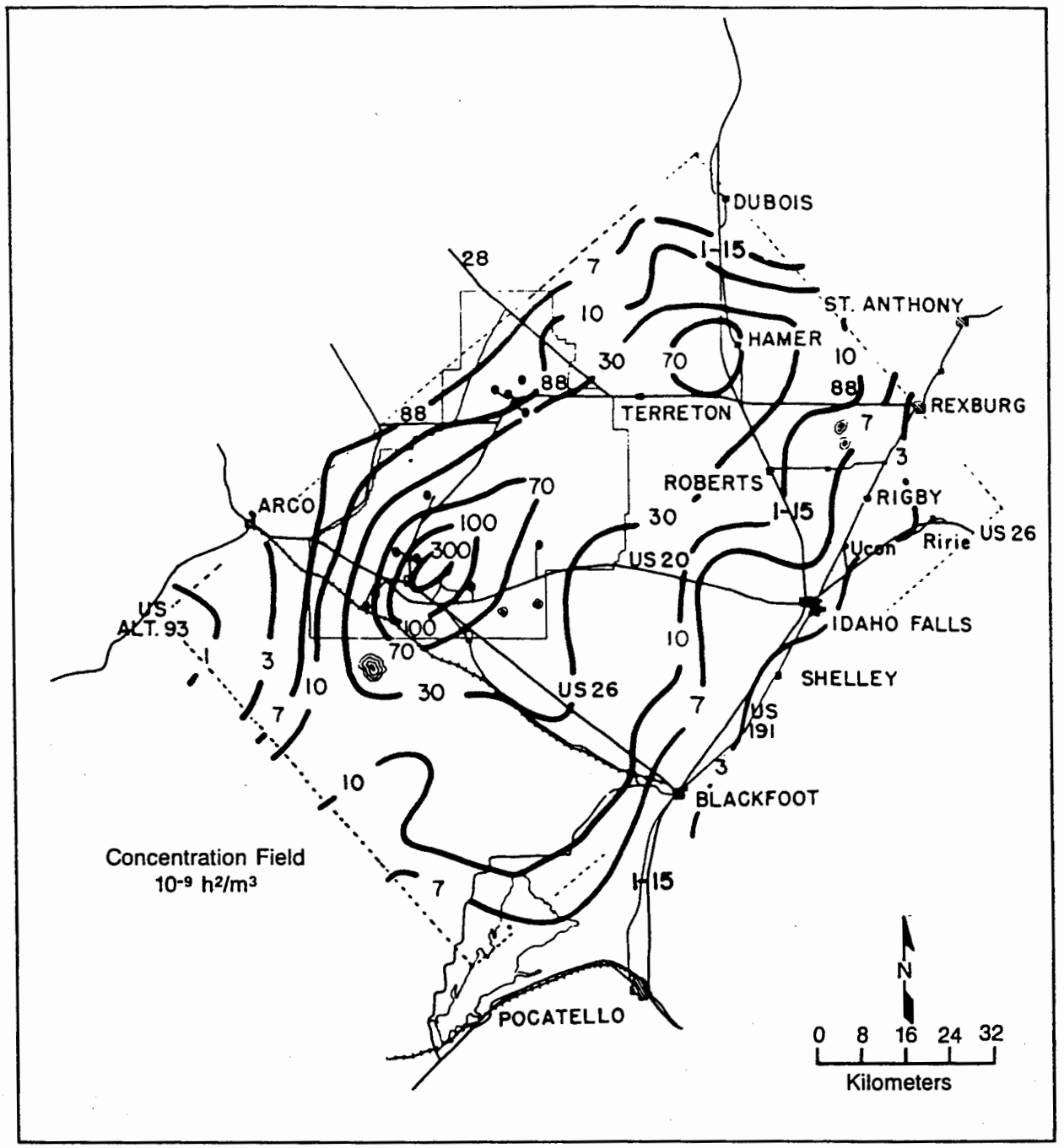

Figure B-16. 1977 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (DOE, 1978).

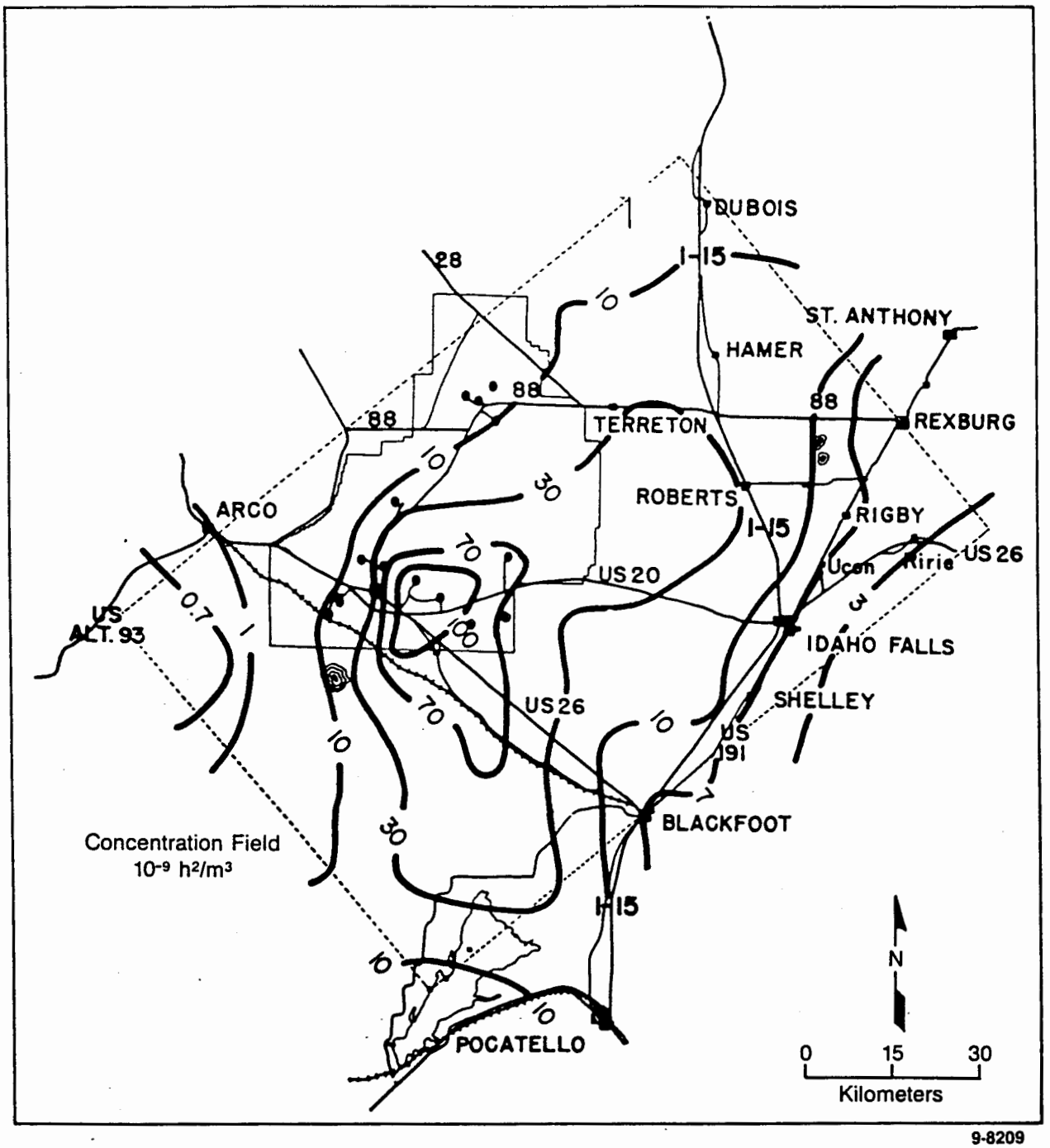

Figure B-17. 1976 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (ERDA, 1977).

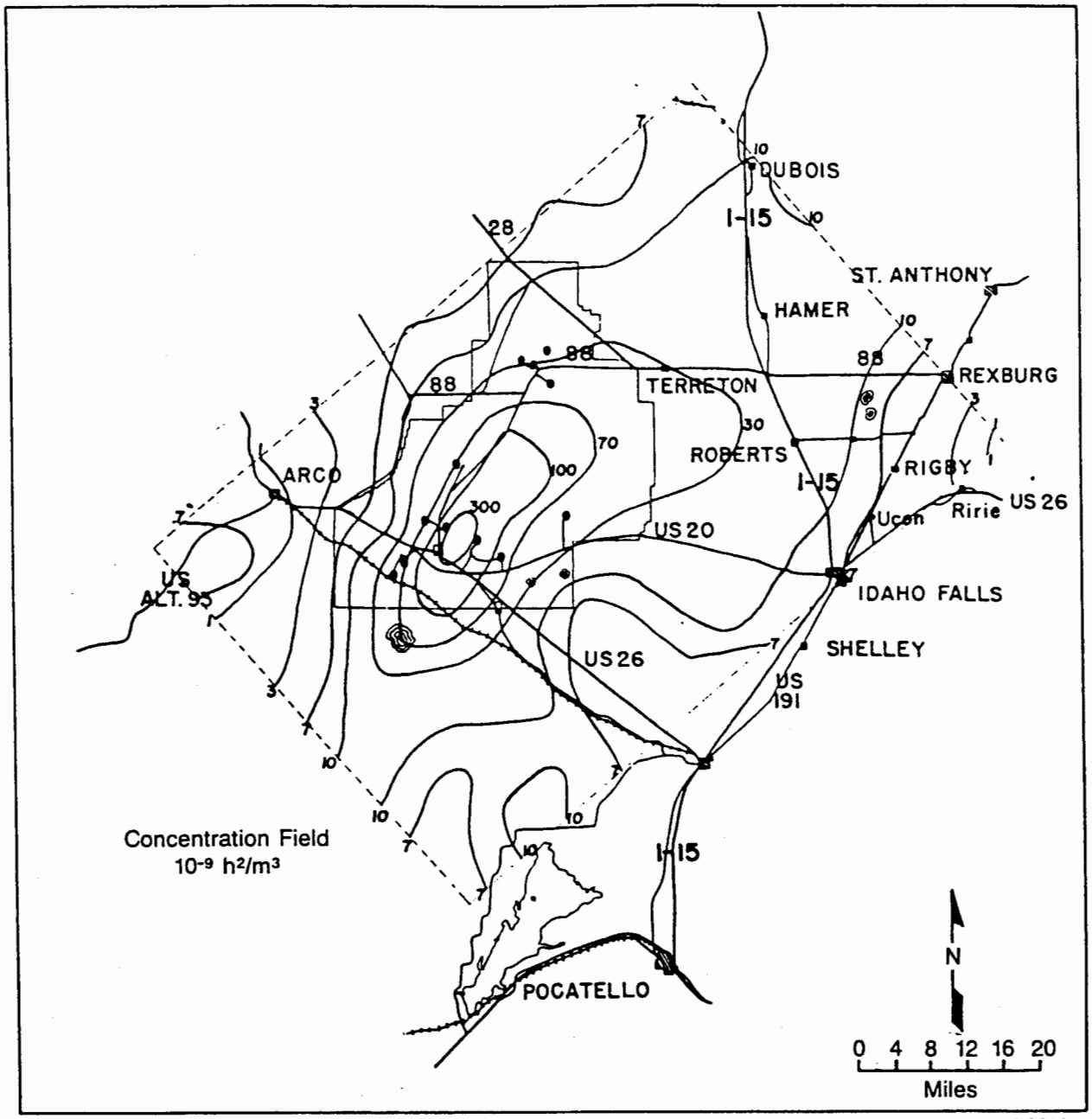

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Figure B-18. 1975 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (ERDA, 1976).

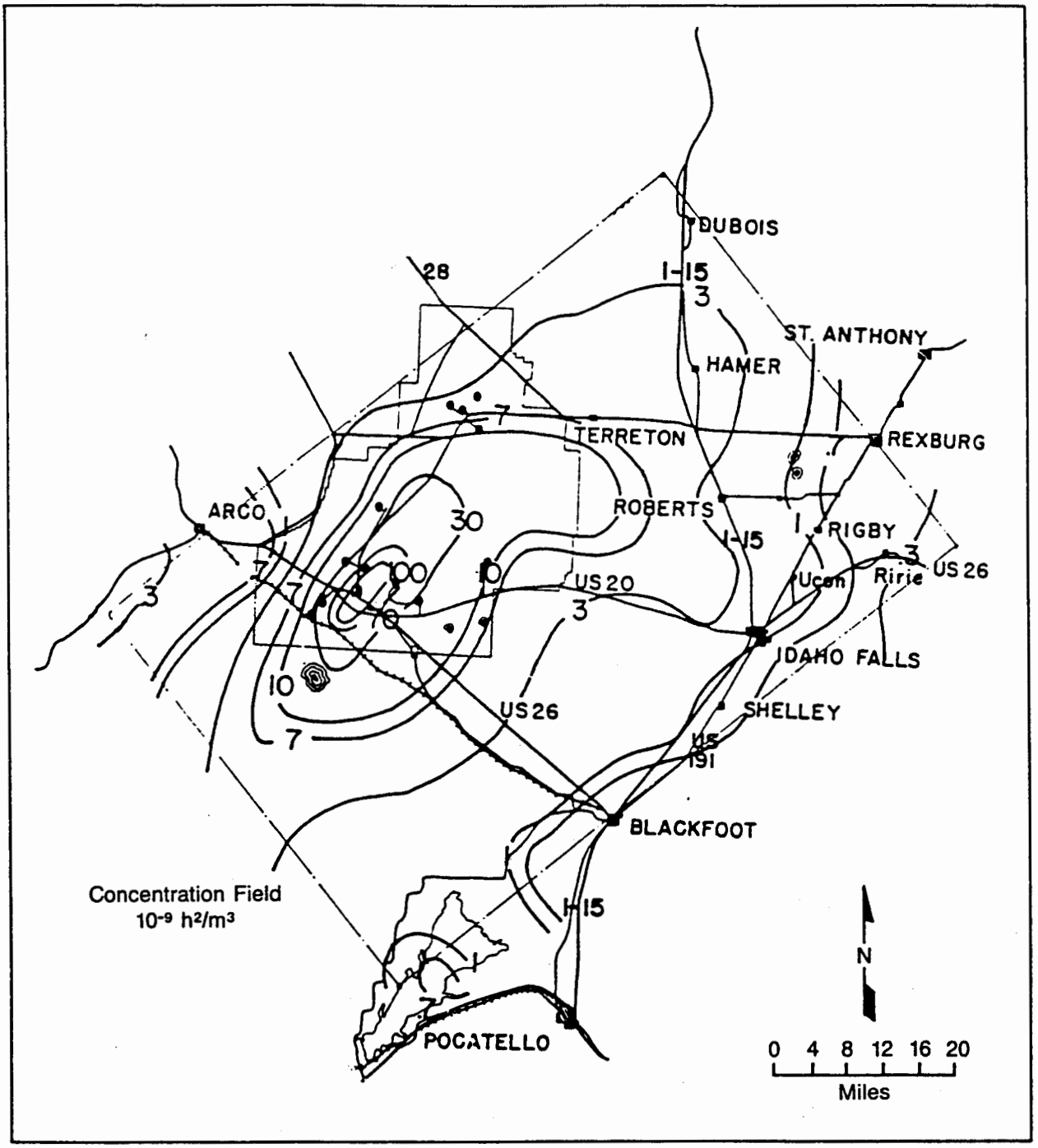

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Figure B-19. 1974 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (ERDA, 1975).

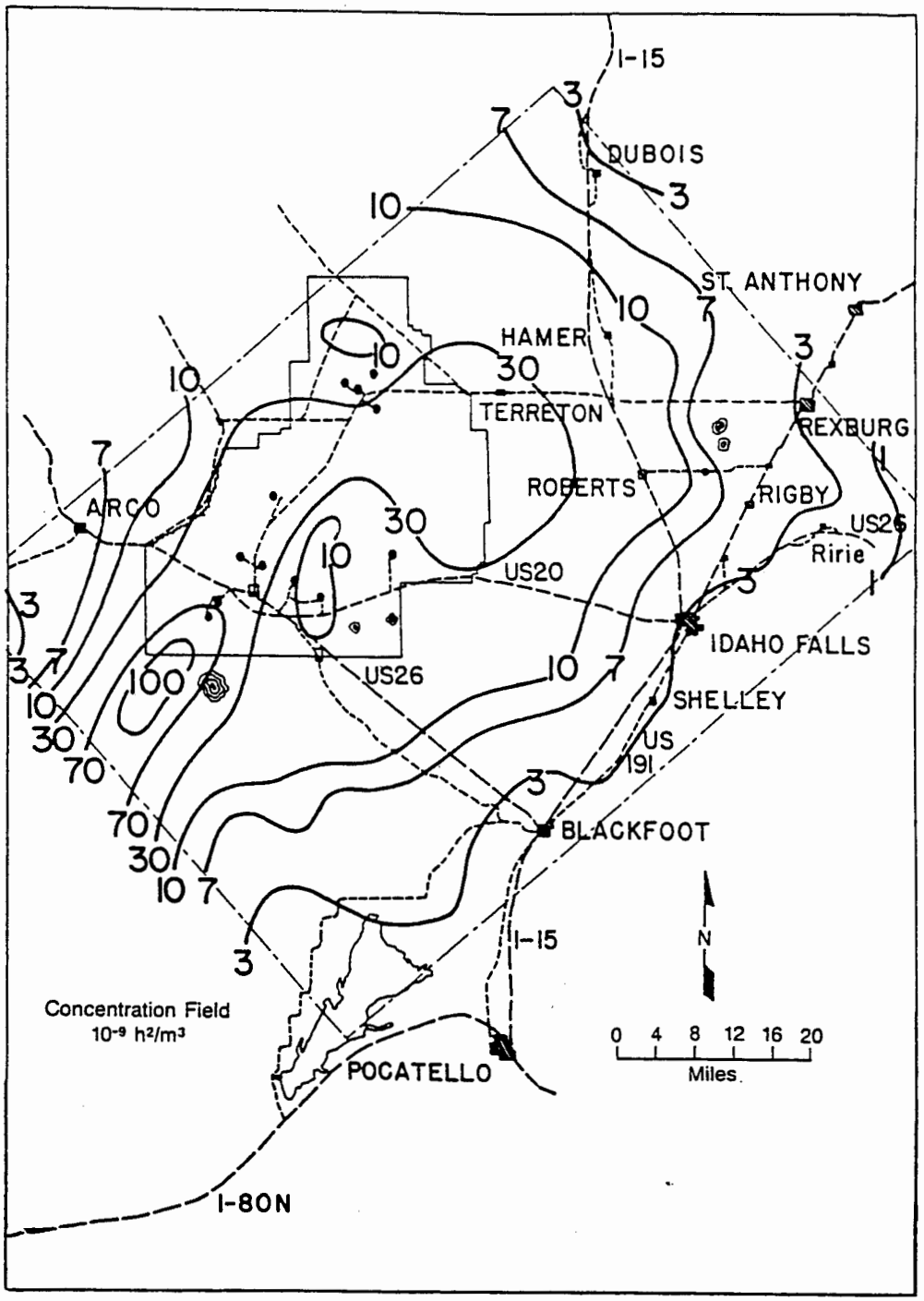


Figure B-20. 1973 average of mesocale dispersion isopleths of air concentrations at ground level, normalized to unit release rate (AEC, 1974).

Table B-1. Meteorological parameters for operational releases

	Maximally Exposed Individual			
Year	Dispersion Coefficient (h^2/m^3)	Location	Distance (km)	Average Wind Speed (km/h)
1989	80E-9	Atomic City	19.1	8.7
1988	90E-9	Terreton	54.0	28.1
1987	44E-9	Atomic City	19.1	13.4
1986	120E-9	Atomic City	19.1	11.1
1985	100E-9	Atomic City	19.1	9.4
1984	45E-9	Atomic City	19.1	12.7
1983	50E-9	Atomic City	19.1	16.4
1982	40E-9	Atomic City	19.1	16.4
1981	75E-9	Atomic City	19.1	16.4
1980	80E-9	Atomic City	19.1	16.4
1979	70E-9	Atomic City	19.1	16.4
1978	50E-9	Atomic City	19.1	16.4
1977	80E-9	13 km WSW of Hamer	63.6	27.3
1976	95E-9	Atomic City	19.1	16.4
1975	50E-9	Atomic City	19.1	16.4
1974	20E-9	Atomic City	19.1	16.4
1973	35E-9	Terreton	49.9	27.3
1972-1952	50E-9	Atomic City	19.1	16.4

Table B-2. Dispersion factors for episodic releases

Date	Description	Highest Dispersion Factor (h/m^3)	Location	Time to INEL Site Boundary (h)	Hours Modeled
9/22/67	CERT No. 22	3.45E-12	Reno Ranch	1.4	1.9
7/6/67	CERT No. 20	6.98E-13	Monteview	1.1	1.9
7/21/66	CERT No. 11	8.97E-13	Building	1.4	2.9
6/14/66	CERT No. 10	2.63E-13	Roberts	1.5	3.9
1/11/66	SNAPTRAN-2	5.72E-12	Building	2.9	5.9
11/22/65	CERT No. 7	6.33E-12	Howe	10.0	25.9
9/14/65	CERT No. 6	6.57E-13	Building	1.0	1.9
6/10/65	CERT No. 5	4.38E-12	Atomic City	0.8	2.9
5/27/65	CERT No. 4	1.17E-13	Mud Lake	7.8	11.9
12/11/64	CERT No. 3	1.96E-11	Roberts	1.9	12.9
9/2/64	CERT No. 2	4.27E-12	Mud Lake	1.2	1.9
4/14/64	SPERT-I, Test No. 3	1.71E-12	2 mi south of Hammer	1.2	2.9
4/1/64	SNAPTRAN-3	3.66E-12	Reno Ranch	0.6	0.9
11/10/63	SPERT-I, Test No. 2	3.53E-14	Cellar	3.9	7.9
5/27/63	CERT No. 1	5.13E-12	Idaho Falls	0.7	39.9
11/5/62	SPERT-I, Test No. 1	8.40E-13	Building	1.1	1.9
1/25/61	1961 ICPP Criticality	9.70E-12	Cerro Grande	2.3	4.9
1/3/61	SL-1 Accident	8.00E-11	Atomic City	1.0	16
3/1-3/30/61	IET #26(B)	3.32E-12	Howe	Var[a]	361
12/23/60-2/28/61	IET #26(A)	6.74E-12	Frenchman's Cabin	Var	826.7
12/1-12/15/60	IET #25(B)	3.90E-12	Cerro Grande	Var	189.2
11/22-11/30/60	IET #25(A)	7.41E-12	Frenchman's Cabin	Var	105
10/26/60	IET #24 (LIME-II)	6.65E-12	Building	0.4	0.9
9/7-10/14/60	IET #23 (FEET #2)	3.37E-12	Monteview	Var	310.4
8/25/60	IET #22 (LIME)	1.30E-12	Monteview	0.5	1.9
6/29-8/6/60	IET #21 (FEET #1)	3.49E-12	Butte City	Var	289
5/14-6/10/60	IET #20	1.75E-12	Blackfoot	Var	371.2
3/1-4/30/60	IET #19(B)	3.14E-12	Roberts	Var	553.8
2/17-2/29/60	IET #19(A)	1.12E-11	Frenchman's Cabin	Var	116.3
1/6-2/5/60	IET #18	5.47E-12	Cerro Grande	Var	275
12/1-12/12/59	IET #17(B)	6.95E-12	Frenchman's Cabin	Var	222
11/2-11/30/59	IET #17(A)	1.10E-11	Cellar	Var	352
10/16/59	1959 ICPP Criticality	2.07E-11	Near Frenchman's Cabin	0.5	2.9
10/9/59	IET #16	4.74E-12	Cerro Grande	1.6	32.9
6/16-6/23/59	IET #15(B)	2.41E-12	Frenchman's Cabin	Var	74.4

Table B-2. (continued)

Date	Description	Highest Dispersion Factor (h/m^3)	Location	Time to INEL Site Boundary (h)	Hours Modeled
6/3-6/15/59	IET #15(A)	9.86E-13	Reno Ranch	Var	223
4/24-5/19/59	IET #14	2.12E-12	Frenchman's Cabin	Var	241.3
11/18/58	IET #13 (HTRE No. 3 Excursion)	4.77E-11	Howe	1.4	5.9
10/29-10/30/58	FECF Filter Break	4.83E-11	Frenchman's Cabin	1.4	5.9
9/26/58	FPFRT-9	3.14E-11	Atomic City	15.4	55.9
9/18/58	FPFRT-8	5.83E-10	Mud Lake	14.1	22.9
9/17/58	FPFRT-7	4.34E-11	Mud Lake	21.9	39.9
9/4/58	FPFRT-6	6.77E-13	Roberts	6.2	52.9
8/27/58	FPFRT-5	4.63E-11	Mud Lake	5.8	16.9
8/14/58	FPFRT-4	9.75E-11	Mud Lake	3.3	27.9
8/6/58	FPFRT-3	8.33E-11	Cellar	10.0	26.9
8/4/58	FPFRT-2	3.05E-12	Blackfoot	2.2	8.9
7/25/58	FPFRT-1	1.90E-12	Cellar	1.4	14.9
5/2/58	IET #12 (BOOT)	2.68E-13	Reno Ranch	1.4	3.9
3/20-4/14/58	IET #11	1.48E-12	Cellar	Var	260
3/1-3/6/58	IET #10(B)	2.72E-12	Cellar	Var	423.5
12/20/57-2/25/58	IET #10(A)	5.32E-12	Howe	Var	150.7
7/31-8/28/57	IET #8	2.79E-12	Monteview	Var	166.5
3/20/57	Fuel Element Burn Test, B	1.58E-10	Birch Creek	2.8	3.9
3/20/57	Fuel Element Burn Test, A	1.01E-12	Reno Ranch	1.9	2.9
12/18/56	IET #6	9.30E-14	Atomic City	5.7	7.9
6/29/56	IET #4(C)	2.50E-12	Monteview	0.3	0.9
5/24-6/26/56	IET #4(B)	1.55E-12	Building	0.7	874
5/1-5/23/56	IET #4(A)	2.24E-12	Building	1.1	545
2/11-2/24/56	IET #3	1.32E-10	Building	0.8	34.6
6/18-7/1/55	NRF S1W Engineering Test	1.40E-11	Building	3.7	349.9
7/22/54	BORAX-I Excursion	7.41E-12	Frenchman's Cabin	0.9	7.9

a. Var = varies throughout the test series. Releases for each run of the test series were modeled and decayed according to the transit time to the INEL Site boundary.

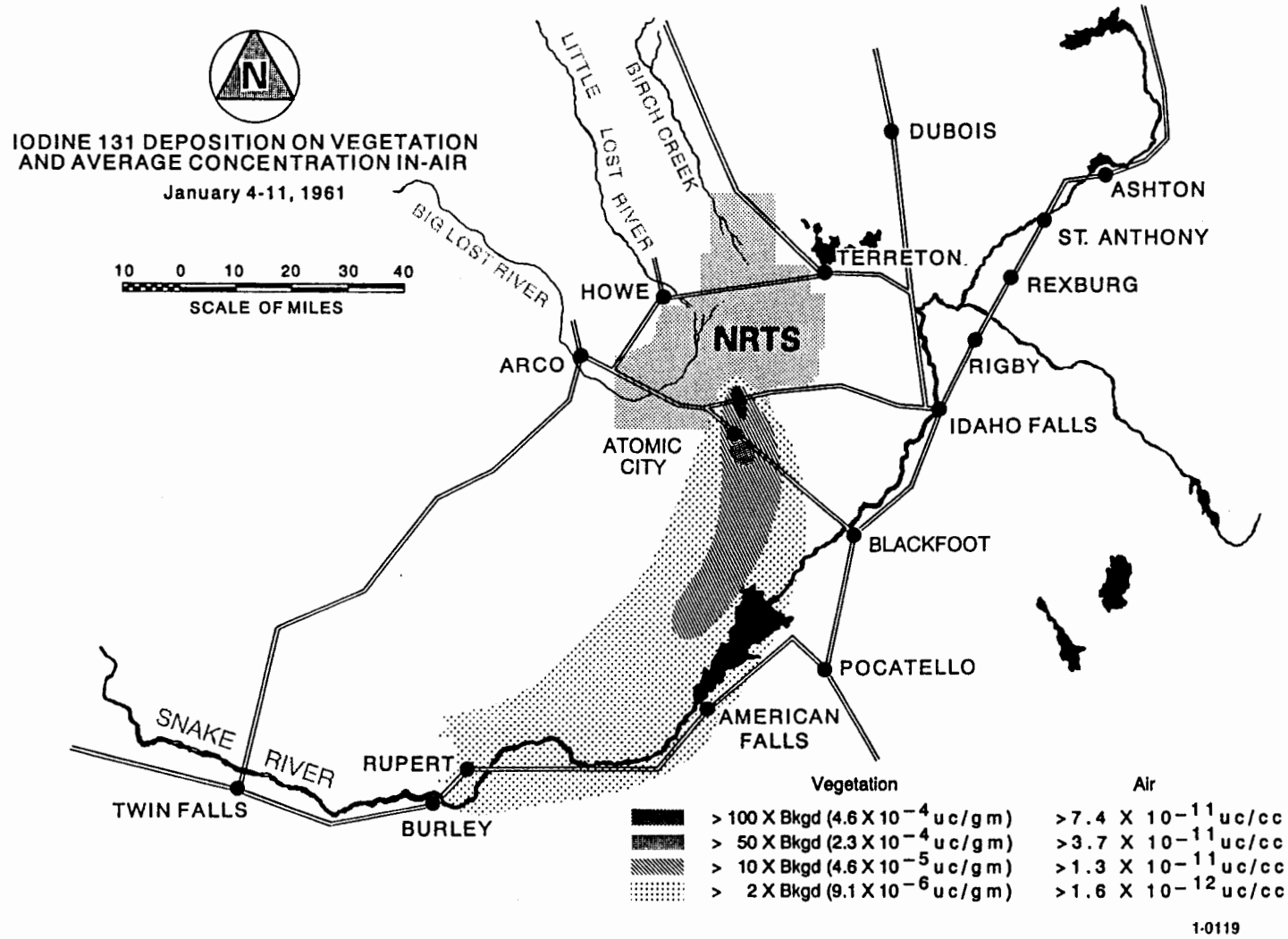
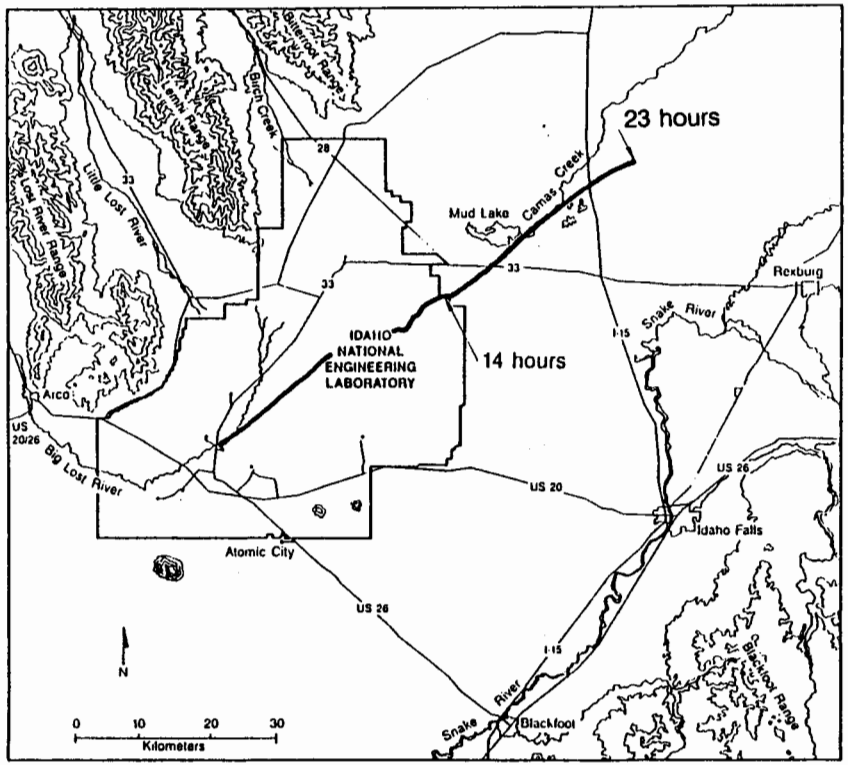
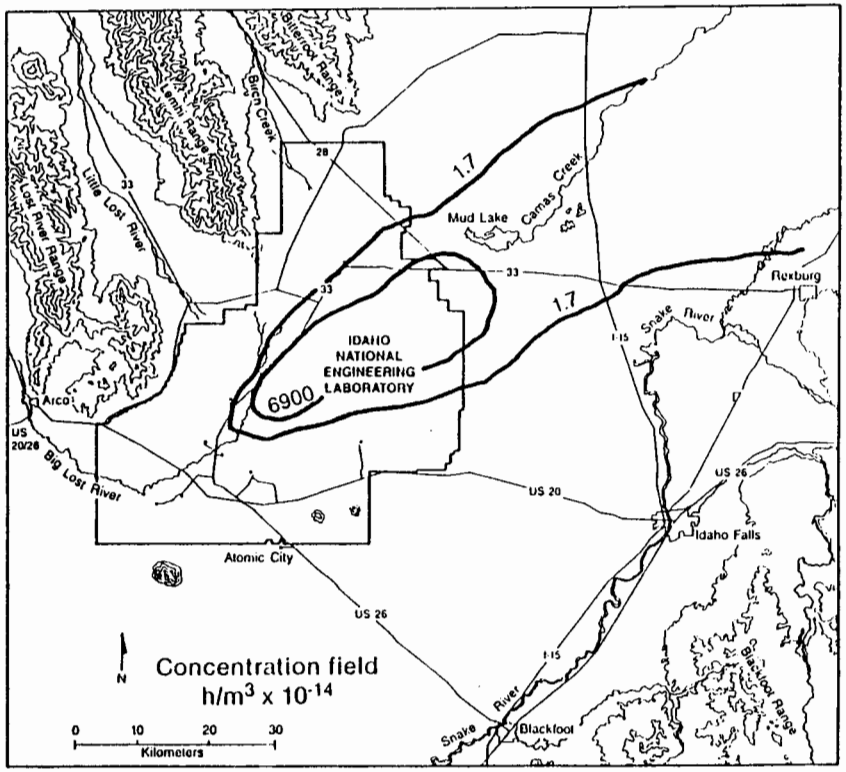


Figure B-21. Dispersion coefficient contours for the SL-1 Accident (redrafted from Horan and Gammill, 1961).



9-8320

Figure B-22. Release trajectory for FPFRT-8 of 9/18/58.



9-8304

Figure B-23. Dispersion coefficient contours for FPFRT-8 of 9/18/58.

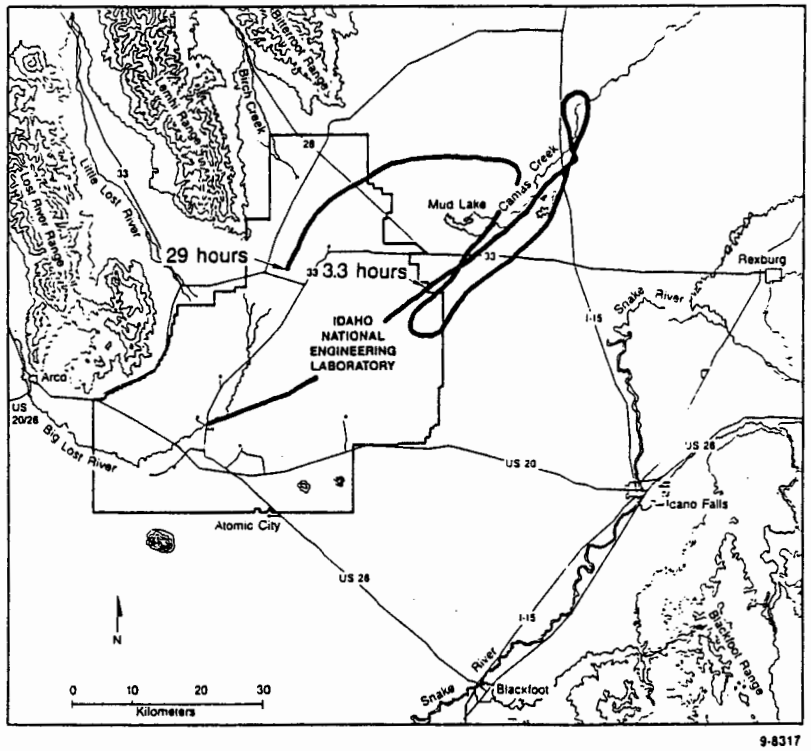


Figure B-24. Release trajectory for FPFRT-4 of 8/14/58.

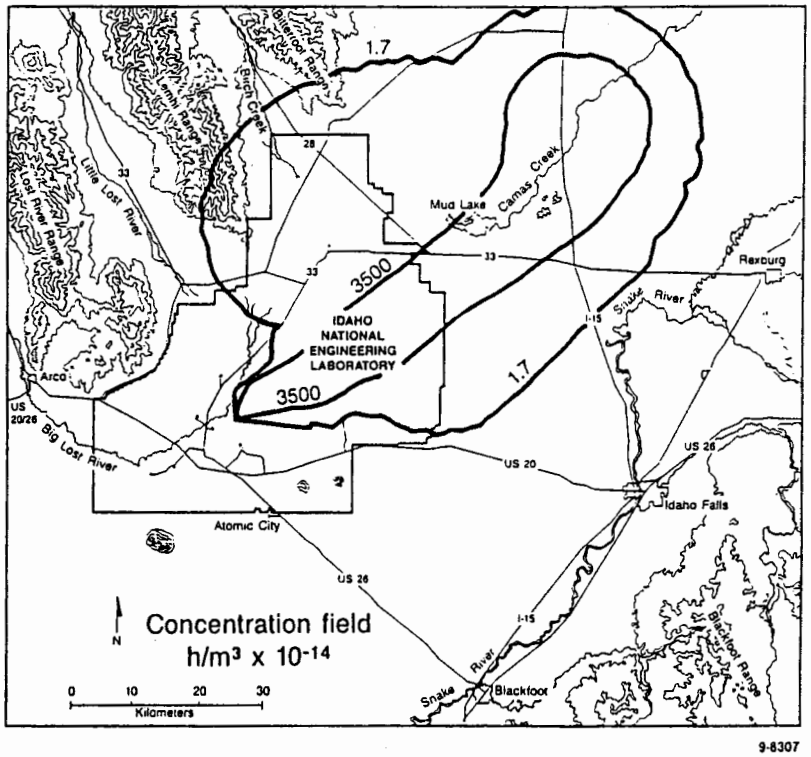


Figure B-25. Dispersion coefficient contours for FPFRT-4 of 8/14/58.

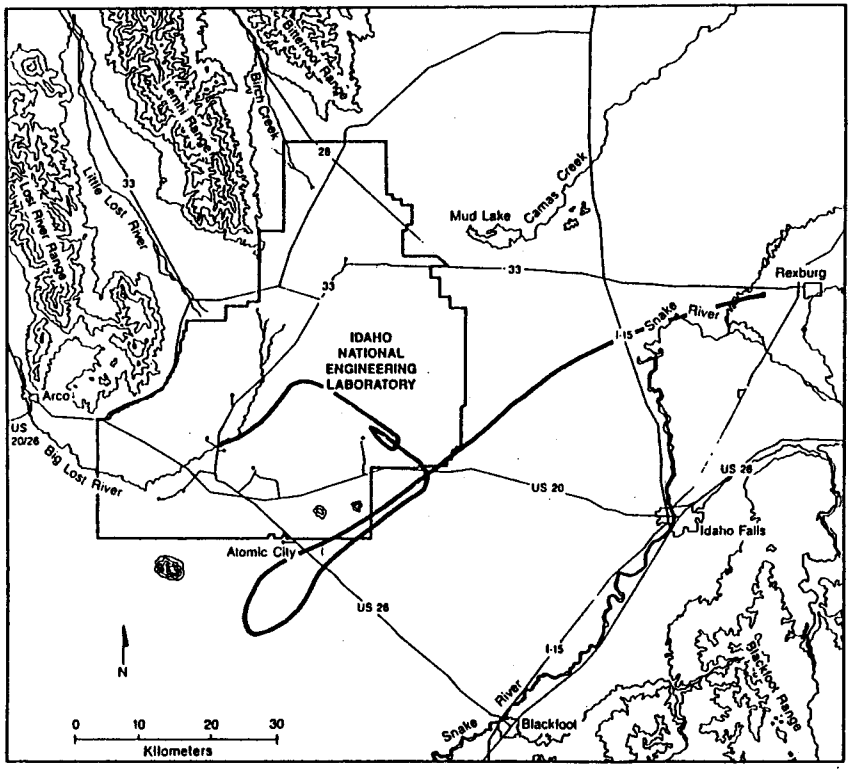


Figure B-26. Release trajectory for FPFRT-3 of 8/6/58.

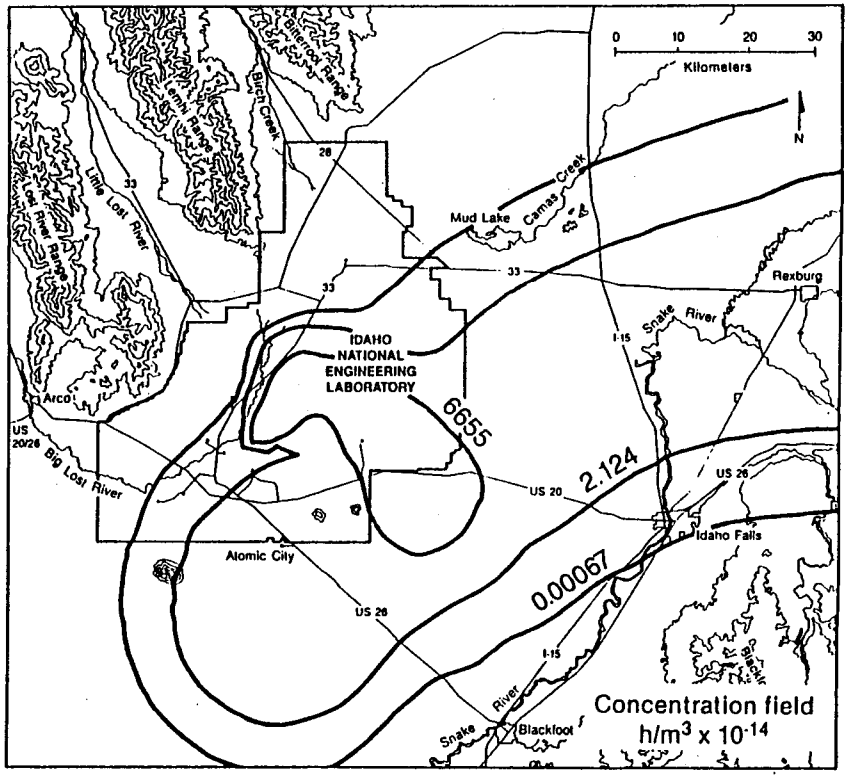


Figure B-27. Dispersion coefficient contours for FPFRT-3 of 8/6/58.

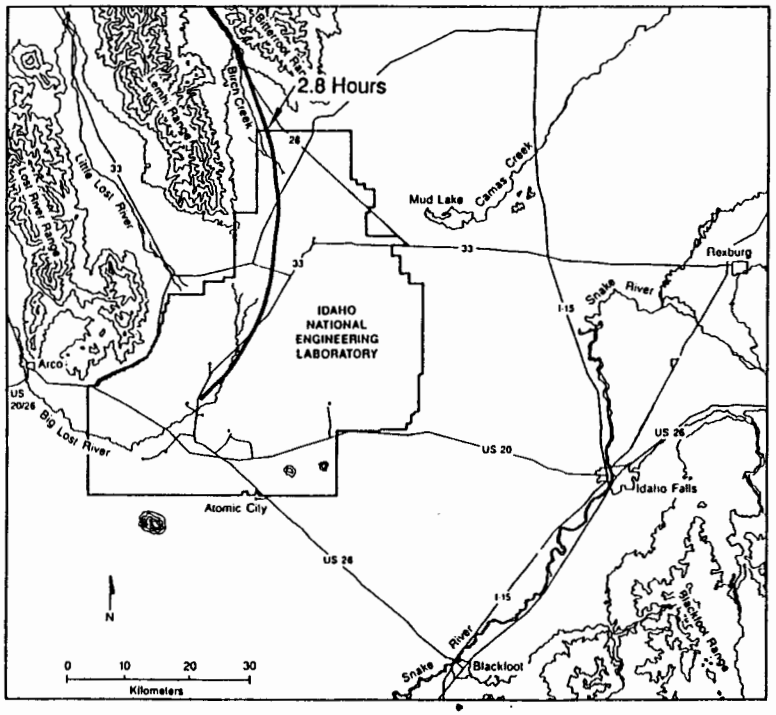


Figure B-28. Release trajectory for Fuel Element Burn Test, B of 3/20/57.

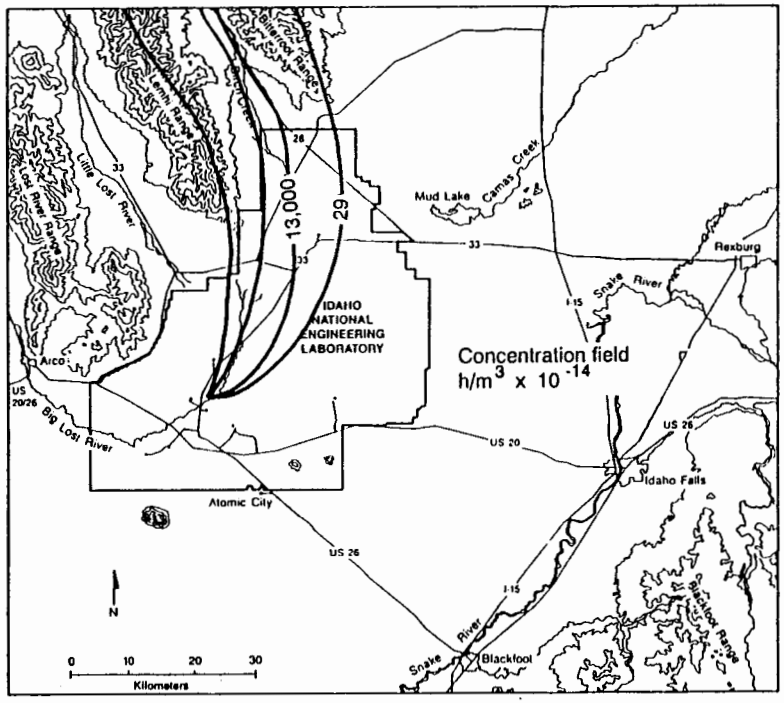


Figure B-29. Dispersion coefficient contours for Fuel Element Burn Test, B of 3/20/57.

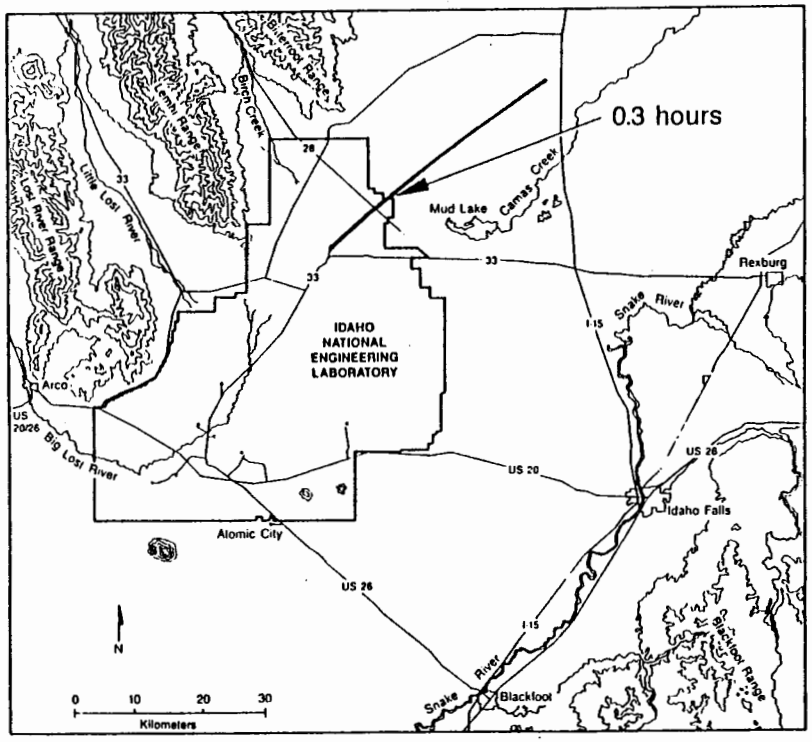


Figure B-30. Release trajectory for IET #4(C) of 6/19/56.

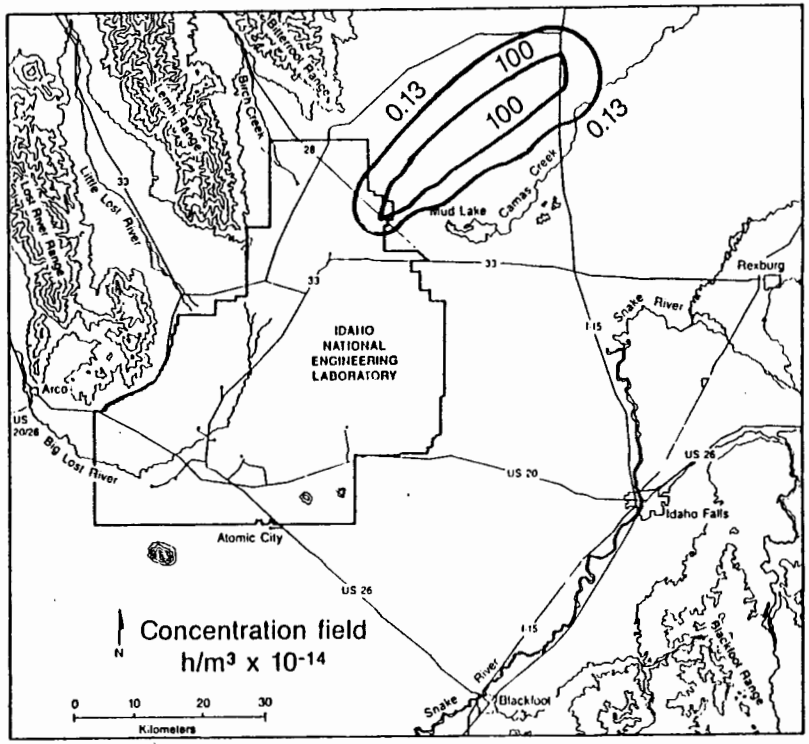


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Figure B-31. Dispersion coefficient contours for IET #4(C) of 6/29/56.

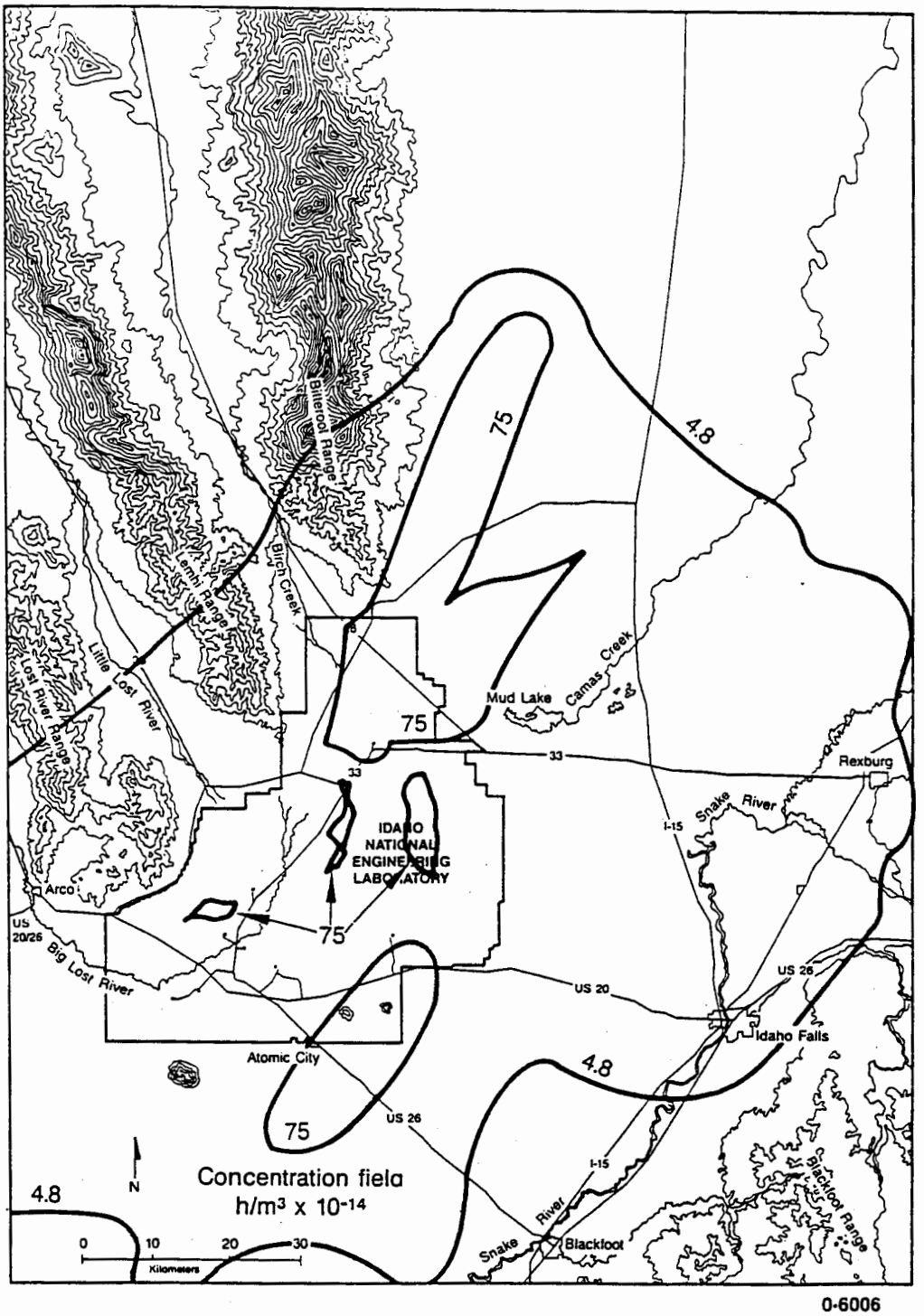

Figure B-32. Dispersion coefficient contours for IET #4(B) of 5/24-6/26/56.

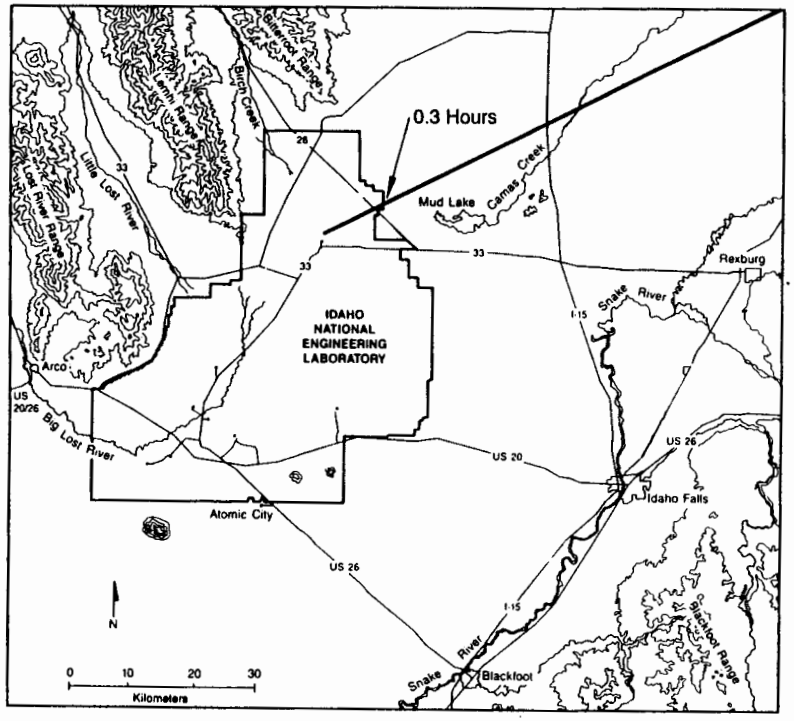

Figure B-33. Release trajectory for IET #3 of 2-11-2/14/56.

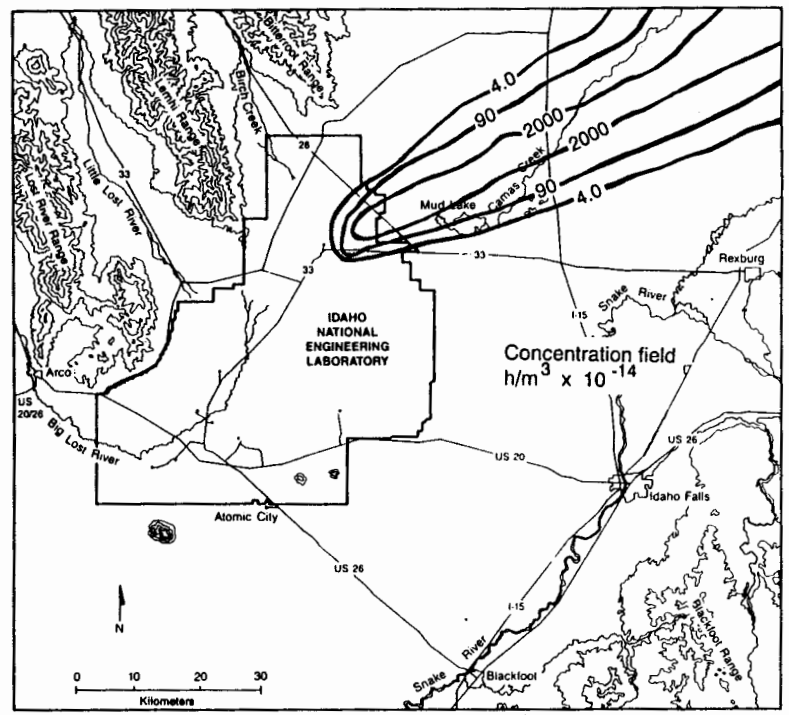

Figure B-34. Dispersion coefficient contours for IET #3 of 2/11-2/24/56.

Table B-3. Dispersion factors for FPFRT-8 of 9/18/58 at 5:24 p.m., 22.9 hours from the start of the release

X	Y	Station	h/m^3
40.50	52.85	Arco	0.00E+00
42.26	53.41	Butte City	0.00E+00
53.92	56.85	Atomic City	0.00E+00
42.20	61.74	Howe	0.00E+00
49.82	72.27	Mud Lake	5.83E-10
45.20	74.20	Monteview	0.00E+00
59.00	76.00	Roberts	0.00E+00
66.28	72.00	Idaho Falls	0.00E+00
68.00	59.15	Blackfoot	0.00E+00
41.23	72.77	Reno Ranch	0.00E+00
40.23	43.95	Craters of the Moon	0.00E+00
51.82	54.55	Cerro Grande	0.00E+00
49.86	51.05	Frenchman's Cabin	0.00E+00
57.50	66.00	Cellar	0.00E+00
46.72	72.90	Building	1.79E-15
34.90	72.40	Birch Creek	0.00E+00

Table B-4. Dispersion factors for FPFRT-4 of 8/14/58 at 10:12 p.m., 27.9 hours from the start of release

X	Y	Station	h/m^3
40.50	52.85	Arco	0.00E+00
42.26	53.41	Butte City	0.00E+00
53.92	56.85	Atomic City	0.00E+00
42.20	61.74	Howe	2.08E-15
49.82	72.27	Mud Lake	9.75E-11
45.20	74.20	Monteview	2.18E-12
59.00	76.00	Roberts	5.67E-16
66.28	72.00	Idaho Falls	0.00E+00
68.00	59.15	Blackfoot	0.00E+00
41.23	72.77	Reno Ranch	4.51E-13
40.23	43.95	Craters of the Moon	0.00E+00
51.82	54.55	Cerro Grande	0.00E+00
49.86	51.05	Frenchman's Cabin	0.00E+00
57.50	66.00	Cellar	3.38E-15
46.72	72.90	Building	1.51E-12
34.90	72.40	Birch Creek	1.01E-17

Table B-5. Dispersion factors for FPFRT-3 of 8/6/58 at 10:06 p.m., 26.9 hours from the start of the release

X	Y	Station	h/m^3
40.50	52.85	Arco	0.00E+00
42.26	53.41	Butte City	0.00E+00
53.92	56.85	Atomic City	3.18E-12
42.20	61.74	Howe	0.00E+00
49.82	72.27	Mud Lake	1.43E-16
45.20	74.20	Monteview	0.00E+00
59.00	76.00	Roberts	1.07E-12
66.28	72.00	Idaho Falls	4.40E-18
68.00	59.15	Blackfoot	0.00E+00
41.23	72.77	Reno Ranch	0.00E+00
40.23	43.95	Craters of the Moon	0.00E+00
51.82	54.55	Cerro Grande	3.97E-13
49.86	51.05	Frenchman's Cabin	3.95E-16
57.50	66.00	Cellar	8.33E-11
46.72	72.90	Building	0.00E+00
34.90	72.40	Birch Creek	0.00E+00

Table B-6. Dispersion factors for the Fuel Element Burn Test, B of 3/20/57 at 10:42 p.m., 3.9 hours from the start of the release

X	Y	Station	h/m^3
40.50	52.85	Arco	0.00E+00
42.26	53.41	Butte City	0.00E+00
53.92	56.85	Atomic City	0.00E+00
42.20	61.74	Howe	0.00E+00
49.82	72.27	Mud Lake	0.00E+00
45.20	74.20	Monteview	0.00E+00
59.00	76.00	Roberts	0.00E+00
66.28	72.00	Idaho Falls	0.00E+00
68.00	59.15	Blackfoot	0.00E+00
41.23	72.77	Reno Ranch	4.79E-12
40.23	43.95	Craters of the Moon	0.00E+00
51.82	54.55	Cerro Grande	0.00E+00
49.86	51.05	Frenchman's Cabin	0.00E+00
57.50	66.00	Cellar	0.00E+00
46.72	72.90	Building	0.00E+00
34.90	72.40	Birch Creek	1.58E-10

Table B-7. Dispersion factors for the IET #4(C) of 6/29/56 at 7:54 p.m., 0.9 hours from the start of the release

X	Y	Station	h/m^3
40.50	52.85	Arco	0.00E+00
42.26	53.41	Butte City	0.00E+00
53.92	56.85	Atomic City	0.00E+00
42.20	61.74	Howe	0.00E+00
49.82	72.27	Mud Lake	0.00E+00
45.20	74.20	Monteview	2.50E-12
59.00	76.00	Roberts	0.00E+00
66.28	72.00	Idaho Falls	0.00E+00
68.00	59.15	Blackfoot	0.00E+00
41.23	72.77	Reno Ranch	0.00E+00
40.23	43.95	Craters of the Moon	0.00E+00
51.82	54.55	Cerro Grande	0.00E+00
49.86	51.05	Frenchman's Cabin	0.00E+00
57.50	66.00	Cellar	0.00E+00
46.72	72.90	Building	5.50E-16
34.90	72.40	Birch Creek	0.00E+00

Table B-8. Dispersion factors for IET #4(B) of 5/24/56-6/26/56 at 10:54 p.m., 873.9 hours from the start of the release[a]

X	Y	Station	h/m^3
40.50	52.85	Arco	2.47E-13
42.26	53.41	Butte City	4.64E-13
53.92	56.85	Atomic City	5.00E-13
42.20	61.74	Howe	1.73E-13
49.82	72.27	Mud Lake	2.53E-13
45.20	74.20	Monteview	1.26E-12
59.00	76.00	Roberts	1.24E-13
66.28	72.00	Idaho Falls	8.15E-14
68.00	59.15	Blackfoot	3.07E-14
41.23	72.77	Reno Ranch	5.69E-13
40.23	43.95	Craters of the Moon	1.54E-13
51.82	54.55	Cerro Grande	9.62E-14
49.86	51.05	Frenchman's Cabin	8.92E-14
57.50	66.00	Cellar	5.17E-13
46.72	72.90	Building	1.55E-12
34.90	72.40	Birch Creek	1.26E-14

a. 95% of puffs reached the INEL Site boundary before 1.7 hours. 95% of puffs reached the INEL Site boundary after 0.2 hours. Average time to the INEL Site boundary for 1289 puffs was 0.7 hours.

Table B-9. Dispersion factors for IET #3 of 2/11-2/24/56, 34.6 hours modeled

X	Y	Station	h/m^3
40.50	52.85	Arco	0.00E+00
42.26	53.41	Butte City	0.00E+00
53.92	56.85	Atomic City	0.00E+00
42.20	61.74	Howe	0.00E+00
49.82	72.27	Mud Lake	9.18E-18
45.20	74.20	Monteview	9.34E-14
59.00	76.00	Roberts	0.00E+00
66.28	72.00	Idaho Falls	0.00E+00
68.00	59.15	Blackfoot	0.00E+00
41.23	72.77	Reno Ranch	0.00E+00
40.23	43.95	Craters of the Moon	0.00E+00
51.82	54.55	Cerro Grande	0.00E+00
49.86	51.05	Frenchman's Cabin	0.00E+00
57.50	66.00	Cellar	0.00E+00
46.72	72.90	Building	1.32E-10
34.90	72.40	Birch Creek	0.00E+00

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APPENDIX C
DOSE CALCULATIONS

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C.1 INTRODUCTION

This evaluation provides a conservative but realistic estimate of offsite radiation doses because of airborne emissions from the Idaho National Engineering Laboratory (INEL) during the operating history of the INEL Site from 1952 to 1989. INEL operations include routine operations of nuclear reactor facilities and the Idaho Chemical Processing Plant (ICPP), for which releases of radioactivity have been reported annually, and a number of short-term or episodic releases, which are treated on a case-by-case basis. The episodic releases include specialized reactor and fuel processing and atmospheric dispersion tests and a small number of accidental releases.

Radiation dose estimates were made using U. S. Nuclear Regulatory Commission (NRC) and U. S. Environmental Protection Agency (EPA) radiation dose assessment methods that apply generic models to calculate dose to hypothetical individuals. Therefore, the estimates reported here should not be taken as representing actual doses received by individuals residing in Southeastern Idaho. While conservative values were chosen wherever information was lacking or assumptions were required, excessive conservatism has been avoided so that these estimates reasonably represent the potential impacts of INEL Site activities. This appendix outlines the methodology used for dose calculations and documents the major assumptions required.

C.2 METHODS

C.2.1 Hypothetically Exposed Individual Scenarios

Dose estimates were made for a hypothetical individual using regulatory assessment procedures that treat (a) external radiation exposure from immersion in contaminated air and exposure to contaminated ground surfaces and (b) internal dose from inhalation and ingestion of radioactivity. To evaluate potential radiation dose, these procedures require that scenarios be defined describing the location and circumstances for the potential radiation dose.

The hypothetical individual receiving the largest dose for each year was determined by evaluating the combined radiation dose for the operational and episodic releases in that year. Doses are reported for three types of inhabited offsite locations where (1) annual average air concentrations from operational releases are greatest for each year, (2) integrated air concentrations are greatest for each episodic release, and (3) the sum of operational and episodic releases are greatest for each year. For most years of INEL Site operation, episodic releases did not contribute significantly to offsite dose. In this case, the location of maximum annual average air concentration from operational releases and the location of the individual receiving the largest dose are the same.

For annual operational releases, the exposed individual is assumed to reside continuously at the nearest inhabited offsite location corresponding to the highest annual average atmospheric concentration of radionuclides. In addition, this individual is assumed to consume milk, fresh produce, and livestock grown while continuously exposed to the same atmospheric concentration. For most years of INEL operation, the location with the highest annual average air concentration was Atomic City. The three exceptions are (1) 1977, when the location was WSW of Hamer; (2) 1973, when the location was at Terreton; and (c) 1988, when the location was at Terreton. Methods for determining the maximum annual average atmospheric concentration for each year of operation are described in Appendix B.

For episodic releases, the exposed individual is assumed to reside at the nearest inhabited offsite location corresponding to the highest integrated air concentration from each release. This individual is assumed to remain at the location long enough to be exposed to the entire integrated air concentration from the release. In general, this length of time is greater than the duration of the release itself because of the effects of winds and atmospheric dispersion. The method for choosing this location and determining the integrated air concentration from episodic releases is described in Appendix B.

C.2.2 Radiation Dose Calculations

C.2.2.1 Dose Conversion Factors

The radiological assessment calculations in this evaluation include four modes of exposure for the hypothetical maximally exposed individual: (1) external exposure from immersion in contaminated air, (2) external exposure from contaminated ground surfaces, (3) internal exposure from inhalation of contaminated air, and (4) internal exposure from ingestion of contaminated foodstuffs. Dose conversion factors are available that relate radiation dose for (a) external dose modes to air and ground surface concentrations and (b) internal dose modes to human intake by inhalation and ingestion.

The units of radiation dose used in this evaluation are those currently recommended by the International Commission on Radiological Protection (ICRP) (ICRP, 1977; ICRP, 1976-1983; ICRP, 1986). Organ doses are reported as 50-year committed dose equivalent. Total doses are reported as effective dose equivalent (EDE) (i.e., the weighted sum of committed dose equivalents to the various tissues of the body). The weights assigned to each organ and tissue reflect their relative sensitivity to radiation damage; therefore, the EDE reflects the total risk from doses to all tissues.

The definition of EDE differs from the older dosimetric unit of whole body dose, which was based on the average energy deposited by radiation over the entire body. This is important because regulatory limits and guidelines

applicable to most of the years treated in this evaluation were expressed in units of the older dosimetric system. Most previously published discussions of radiation dose estimates are also expressed in slightly different units than those currently used. For most purposes of comparison, the EDE can be used interchangeably with the whole body dose equivalent and serves equally well for comparison with historical regulatory limits and guidelines. Organ doses are reported as the 50-year committed dose equivalent and are directly comparable to the critical organ regulatory limits and guidelines applicable in past years.

Dose equivalent is reported in units of rem (or mrem), and radioactivity is reported in units of Ci (or μCi). Dose conversion factors for immersion are expressed in mrem/yr per $\mu Ci/m^3$, and dose conversion factors for exposure to ground surfaces are expressed in mrem/yr per $\mu Ci/m^2$. Internal dose conversion factors are expressed in rem/μCi intake.

Three age groups were treated in this evaluation: (1) adults, (2) children (10-year old), and (3) infants (0-12 months old). These age groups were chosen because they demonstrate the largest differences in dose to specific organs from radionuclides of concern at the INEL Site. Differences in dose estimates among age groups arise primarily from two considerations: (1) dose conversion factors for inhaled and ingested radioactivity vary with age, and (2) diets differ with age. Age-specific differences in dose conversion factors are presented here. Differences in dietary assumptions are presented in Section C.2.2.6.

All dose conversion factors for the adult age group were taken from compilations prepared for use by the U. S. Department of Energy (DOE) to calculate dose to the public (DOE, 1988a; DOE, 1988b). These dose conversion factors are based on the most recent recommendations of the ICRP (ICRP, 1977; ICRP, 1976-1983; ICRP, 1986). When radioactive daughter products contributed significantly to the external dose, the parent and daughter radionuclides were assumed to be in equilibrium and the daughter's contribution to the immersion or ground deposition doses were included with the dose conversion factor of the parent radionuclide (see Section A.2.1). For internal exposure pathways, no initial intake of radioactivity was assumed from daughter radionuclides in

parent-daughter decay chains. The ingrowth of the radioactive daughters after an internal intake is accounted for in the dose conversion factors for internal doses. Tables C-1 through C-4 list the dose factors used for estimating EDE and organ doses to adults from all four dose modes of exposure.

For calculating inhalation and ingestion doses, several dose conversion factors are available for each radionuclide. These represent the metabolic behavior of different chemical forms. Because the chemical forms of most releases at the INEL Site are not known, inhalation dose conversion factors were chosen to maximize estimates of EDE rather than dose to any specific organ.

Exposure to tritium (H-3) represents a special case. Dose conversion factors are available for two chemical forms of tritium: tritiated hydrogen gas (HT) and tritiated water (HTO). The dose conversion factors for HTO are significantly larger than for HT. All releases of tritium were assumed to be in the form of HTO, and the corresponding dose conversion factors were used.

Dose rate conversion factors for external exposure from contaminated air and ground surfaces do not vary significantly among age groups. Therefore, the values provided for adults were used for all age groups. Dose conversion factors for internal dose from inhalation and ingestion of radioactivity do show large differences that depend on variations in metabolism and organ size with age. Reliable age-specific dose conversion factors using current ICRP recommended dosimetric methods are not available for many radionuclides. However, recent ICRP studies provide age-specific factors for ingestion and inhalation of the radionuclides that contribute most of the dose from INEL releases (ICRP, 1989). Where these values were not available, adult dose conversion factors were used. The age-specific inhalation dose conversion factors for children and infants are listed in Table C-5 and those for ingestion appear in Table C-6.

C.2.2.2 Air and Ground Concentrations

Air immersion and inhalation doses are proportional to total time integrated air concentration at the receptor point. Doses from ground surfaces and from ingestion of radioactivity are proportional to total time

Table C-1. Dose conversion factors for dose equivalent from air immersion (all age groups) (mrem/y per $\mu Ci/m^3$)[a]

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	2.91E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
N-13	5.11E+03	1.04E+04	5.53E+03	4.45E+03	4.15E+03	5.21E+03	4.56E+03
Na-24	2.39E+04	3.50E+04	2.54E+04	2.13E+04	2.03E+04	2.22E+04	2.17E+04
Ar-41	6.63E+03	1.22E+04	7.34E+03	5.89E+03	5.57E+03	6.29E+03	5.94E+03
Sc-46	1.05E+04	1.46E+04	1.17E+04	9.30E+03	8.74E+03	1.01E+04	9.41E+03
Cr-51	1.58E+02	2.05E+02	1.71E+02	1.35E+02	1.25E+02	1.72E+02	1.39E+02
Mn-54	4.37E+03	5.91E+03	4.86E+03	3.85E+03	3.61E+03	4.25E+03	3.91E+03
Mn-56	9.31E+03	1.88E+04	1.02E+04	8.29E+03	7.83E+03	8.87E+03	8.40E+03
Co-58	5.05E+03	6.99E+03	5.60E+03	4.45E+03	4.17E+03	4.96E+03	4.53E+03
Co-60	1.30E+04	1.77E+04	1.44E+04	1.15E+04	1.09E+04	1.23E+04	1.16E+04
As-76	2.21E+03	1.14E+04	2.42E+03	1.94E+03	1.82E+03	2.19E+03	1.98E+03
Br-82	1.37E+04	1.90E+04	1.51E+04	1.21E+04	1.13E+04	1.34E+04	1.22E+04
Br-84	1.02E+04	2.34E+04	1.09E+04	9.07E+03	8.64E+03	9.59E+03	9.29E+03
Kr-85	1.12E+01	1.58E+03	1.21E+01	9.75E+00	9.08E+00	1.14E+01	9.97E+00
Kr-85m	8.17E+02	2.62E+03	9.03E+02	6.73E+02	6.16E+02	9.90E+02	6.46E+02
Kr-87	4.47E+03	1.65E+04	4.77E+03	3.95E+03	3.75E+03	4.29E+03	4.04E+03
Kr-88+D	1.53E+04	4.18E+04	1.64E+04	1.37E+04	1.30E+04	1.45E+04	1.39E+04
Kr-89	1.02E+04	2.43E+04	1.09E+04	9.01E+03	8.55E+03	9.69E+03	9.19E+03
Kr-90	6.73E+03	1.94E+04	7.37E+03	5.94E+03	5.60E+03	6.56E+03	5.99E+03
Rb-89	1.14E+04	2.31E+04	1.24E+04	1.01E+04	9.58E+03	1.08E+04	1.03E+04
Rb-90	1.29E+04	3.25E+04	1.35E+04	1.13E+04	1.09E+04	1.19E+04	1.18E+04
Sr-89	7.17E-01	4.38E+03	8.01E-01	6.34E-01	5.95E-01	6.94E-01	6.43E-01
Sr-90+D	0.00E+00	8.49E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	5.28E+03	1.22E+04	5.83E+03	4.64E+03	4.34E+03	5.18E+03	4.71E+03
Sr-92	6.93E+03	1.04E+04	7.64E+03	6.16E+03	5.82E+03	6.58E+03	6.22E+03
Y-91	1.87E+01	4.57E+03	2.08E+01	1.66E+01	1.57E+01	1.78E+01	1.68E+01
Y-91m	2.68E+03	3.79E+03	2.92E+03	2.34E+03	2.18E+03	2.70E+03	2.39E+03
Y-92	1.32E+03	1.35E+04	1.46E+03	1.16E+03	1.09E+03	1.27E+03	1.18E+03
Y-93	4.82E+02	1.01E+04	5.25E+02	4.25E+02	4.01E+02	4.73E+02	4.32E+02
Zr-95+D	1.25E+04	1.74E+04	1.39E+04	1.10E+04	1.03E+04	1.23E+04	1.12E+04
Zr-97+D	8.21E+03	2.00E+04	9.04E+03	7.22E+03	6.75E+03	8.09E+03	7.34E+03
Nb-95	3.97E+03	5.40E+03	4.41E+03	3.50E+03	3.28E+03	3.89E+03	3.56E+03
Nb-96	1.28E+04	1.88E+04	1.42E+04	1.13E+04	1.06E+04	1.25E+04	1.14E+04
Mo-99+D	7.99E+02	4.01E+03	8.85E+02	6.98E+02	6.51E+02	8.08E+02	7.06E+02
Tc-99m	6.63E+02	8.40E+02	7.41E+02	5.38E+02	4.88E+02	8.43E+02	4.91E+02
Ru-103+D	2.42E+03	3.39E+03	2.62E+03	2.11E+03	1.96E+03	2.47E+03	2.16E+03
Ru-105+D	4.06E+03	8.28E+03	4.47E+03	3.55E+03	3.32E+03	4.07E+03	3.61E+03
Ru-106+D	1.05E+03	1.28E+04	1.15E+03	9.22E+02	8.61E+02	1.05E+03	9.41E+02
Sb-125+D	2.18E+03	3.43E+03	2.36E+03	1.87E+03	1.74E+03	2.22E+03	1.89E+03
Sb-129+D	7.81E+03	1.72E+04	8.62E+03	6.89E+03	6.47E+03	7.60E+03	6.99E+03
Te-131	2.16E+03	8.15E+03	2.38E+03	1.86E+03	1.73E+03	2.28E+03	1.85E+03
Te-131m+D	7.93E+03	1.27E+04	8.77E+03	6.96E+03	6.53E+03	7.82E+03	7.04E+03
Te-132+D	1.33E+04	2.17E+04	1.47E+04	1.17E+04	1.09E+04	1.32E+04	1.18E+04
Te-133m+D	1.25E+04	2.31E+04	1.38E+04	1.10E+04	1.03E+04	1.22E+04	1.11E+04
Te-134	4.44E+03	6.42E+03	4.88E+03	3.84E+03	3.58E+03	4.58E+03	3.89E+03
I-129	4.17E+01	1.16E+02	4.15E+01	2.06E+01	1.65E+01	3.62E+01	7.82E+00
I-131	1.91E+03	3.52E+03	2.07E+03	1.64E+03	1.52E+03	2.02E+03	1.69E+03
I-132	1.19E+04	1.95E+04	1.31E+04	1.05E+04	9.81E+03	1.16E+04	1.06E+04
I-133	3.07E+03	6.97E+03	3.35E+03	2.69E+03	2.51E+03	3.08E+03	2.74E+03
I-134	1.38E+04	2.31E+04	1.53E+04	1.22E+04	1.14E+04	1.34E+04	1.23E+04
I-135	8.34E+03	1.37E+04	9.17E+03	7.41E+03	7.00E+03	7.95E+03	7.49E+03
Xe-129m	1.17E+02	9.62E+02	1.20E+02	7.29E+01	6.26E+01	1.15E+02	5.18E+01
Xe-131m	4.29E+01	5.61E+02	4.41E+01	2.61E+01	2.21E+01	4.23E+01	1.77E+01
Xe-133	1.76E+02	5.81E+02	2.01E+02	1.30E+02	1.16E+02	2.30E+02	8.49E+01
Xe-133m	1.49E+02	1.21E+03	1.59E+02	1.15E+02	1.04E+02	1.62E+02	1.08E+02
Xe-135	1.25E+03	3.64E+03	1.36E+03	1.05E+03	9.72E+02	1.40E+03	1.07E+03
Xe-135m	2.15E+03	3.57E+03	2.33E+03	1.87E+03	1.74E+03	2.18E+03	1.91E+03
Xe-137	9.55E+02	1.58E+04	1.03E+03	8.35E+02	7.80E+02	9.65E+02	8.54E+02
Xe-138	6.27E+03	1.29E+04	6.75E+03	5.56E+03	5.27E+03	6.05E+03	5.64E+03

Table C-1. (continued)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Xe-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	8.01E+03	1.17E+04	8.83E+03	7.04E+03	6.58E+03	7.91E+03	7.17E+03
Cs-136	1.13E+04	1.56E+04	1.25E+04	9.91E+03	9.29E+03	1.11E+04	1.00E+04
Cs-137+D	2.89E+03	5.23E+03	3.19E+03	2.54E+03	2.37E+03	2.88E+03	2.59E+03
Cs-138	1.27E+04	2.66E+04	1.39E+04	1.13E+04	1.07E+04	1.21E+04	1.15E+04
Cs-139	1.67E+03	1.57E+04	1.81E+03	1.49E+03	1.41E+03	1.57E+03	1.51E+03
Ba-139	1.80E+02	7.20E+03	1.99E+02	1.49E+02	1.37E+02	2.13E+02	1.43E+02
Ba-140+D	1.51E+04	2.63E+04	1.64E+04	1.33E+04	1.26E+04	1.45E+04	1.36E+04
Ba-141	4.59E+03	1.26E+04	5.03E+03	4.00E+03	3.74E+03	4.70E+03	4.06E+03
Ba-142	4.67E+03	9.48E+03	5.18E+03	4.10E+03	3.85E+03	4.61E+03	4.14E+03
La-141	2.23E+02	7.70E+03	2.45E+02	1.98E+02	1.88E+02	2.11E+02	2.00E+02
La-142	1.57E+04	2.70E+04	1.67E+04	1.40E+04	1.33E+04	1.47E+04	1.42E+04
Ce-141	3.86E+02	1.19E+03	4.31E+02	3.11E+02	2.82E+02	4.85E+02	2.82E+02
Ce-143	1.32E+03	4.66E+03	1.44E+03	1.11E+03	1.03E+03	1.39E+03	1.11E+03
Ce-144+D	2.67E+02	1.01E+04	2.92E+02	2.28E+02	2.14E+02	2.84E+02	2.21E+02
Pr-143	4.62E-05	2.07E+03	5.11E-05	4.06E-05	3.80E-05	4.53E-05	4.13E-05
Pr-144	1.79E+02	9.87E+03	1.93E+02	1.59E+02	1.51E+02	1.70E+02	1.62E+02
Pm-147	1.83E-02	9.59E+01	2.06E-02	1.47E-02	1.33E-02	2.39E-02	1.27E-02
Eu-154	6.50E+03	1.01E+04	7.22E+03	5.72E+03	5.37E+03	6.37E+03	5.75E+03
Eu-155	2.87E+02	4.07E+02	3.30E+02	2.22E+02	2.00E+02	3.85E+02	1.61E+02
Ta-182	6.68E+03	9.53E+03	7.44E+03	5.87E+03	5.52E+03	6.58E+03	5.83E+03
Ir-192	4.11E+03	6.55E+03	4.45E+03	3.53E+03	3.27E+03	4.37E+03	3.61E+03
Hg-203	1.15E+03	1.80E+03	1.25E+03	9.66E+02	8.92E+02	1.29E+03	9.76E+02
Th-232	4.89E+03	9.18E+03	5.42E+03	4.30E+03	4.04E+03	4.80E+03	4.35E+03
Np-237+D	1.19E+03	2.19E+03	1.30E+03	9.83E+02	9.04E+02	1.36E+03	9.51E+02
U-234	7.65E-01	3.27E+00	6.08E-01	4.11E-01	3.46E-01	7.09E-01	2.76E-01
U-235+D	8.30E+02	1.26E+03	9.15E+02	6.74E+02	6.15E+02	1.01E+03	6.37E+02
U-238+D	9.90E+01	6.58E+03	1.12E+02	8.30E+01	7.66E+01	1.12E+02	7.41E+01
Pu-238	4.41E-01	3.06E+00	1.71E-01	1.18E-01	7.16E-02	2.09E-01	5.31E-02
Pu-239	4.11E-01	1.45E+00	3.39E-01	2.42E-01	2.05E-01	4.01E-01	1.95E-01
Pu-240	4.32E-01	2.93E+00	1.77E-01	1.21E-01	7.59E-02	2.15E-01	5.45E-02
Pu-239/240	4.11E-01	1.45E+00	3.39E-01	2.42E-01	2.05E-01	4.01E-01	1.95E-01

a. Dose conversion factors are from DOE (1988a) (bone surface = skeleton).

Table C-2. Dose conversion factors for dose equivalent from ground surfaces (mrem/y per $\mu Ci/m^2$)[a]

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
N-13	1.05E+02	6.03E+02	1.14E+02	9.17E+01	8.55E+01	1.07E+02	9.39E+01
Na-24	3.56E+02	1.00E+03	3.78E+02	3.16E+02	3.02E+02	3.31E+02	3.23E+02
Ar-41	1.15E+02	5.80E+02	1.28E+02	1.02E+02	9.68E+01	1.09E+02	1.03E+02
Sc-46	1.94E+02	2.62E+02	2.16E+02	1.72E+02	1.61E+02	1.86E+02	1.74E+02
Cr-51	3.41E+00	4.53E+00	3.67E+00	2.89E+00	2.68E+00	3.69E+00	2.99E+00
Mn-54	8.41E+01	1.14E+02	9.36E+01	7.42E+01	6.95E+01	8.18E+01	7.53E+01
Mn-56	1.60E+02	1.06E+03	1.75E+02	1.43E+02	1.35E+02	1.53E+02	1.45E+02
Co-58	9.87E+01	1.37E+02	1.09E+02	8.69E+01	8.13E+01	9.69E+01	8.83E+01
Co-60	2.27E+02	3.05E+02	2.52E+02	2.02E+02	1.91E+02	2.16E+02	2.04E+02
As-76	4.27E+01	1.20E+03	4.67E+01	3.75E+01	3.51E+01	4.23E+01	3.82E+01
Br-82	2.60E+02	3.56E+02	2.87E+02	2.29E+02	2.15E+02	2.54E+02	2.33E+02
Br-84	1.59E+02	1.53E+03	1.71E+02	1.41E+02	1.34E+02	1.50E+02	1.44E+02
Kr-85	2.30E-01	1.42E+02	2.49E-01	2.01E-01	1.87E-01	2.34E-01	2.05E-01
Kr-85m	1.80E+01	1.72E+02	1.99E+01	1.48E+01	1.35E+01	2.18E+01	1.42E+01
Kr-87	7.40E+01	1.52E+03	7.92E+01	6.54E+01	6.18E+01	7.18E+01	6.68E+01
Kr-88+D	2.42E+02	3.10E+03	2.59E+02	2.15E+02	2.05E+02	2.28E+02	2.18E+02
Kr-89	1.67E+02	1.68E+03	1.80E+02	1.48E+02	1.40E+02	1.60E+02	1.51E+02
Kr-90	1.20E+02	1.56E+03	1.32E+02	1.06E+02	9.96E+01	1.18E+02	1.07E+02
Rb-89	1.90E+02	1.32E+03	2.08E+02	1.69E+02	1.60E+02	1.81E+02	1.72E+02
Rb-90	1.76E+02	2.33E+03	1.85E+02	1.55E+02	1.48E+02	1.63E+02	1.61E+02
Sr-89	1.36E-02	5.89E+02	1.51E-02	1.20E-02	1.13E-02	1.31E-02	1.22E-02
Sr-90+D	0.00E+00	1.07E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.02E+02	8.09E+02	1.13E+02	8.98E+01	8.41E+01	1.00E+02	9.13E+01
Sr-92	1.19E+02	2.28E+02	1.32E+02	1.06E+02	1.00E+02	1.13E+02	1.07E+02
Y-91	3.31E-01	6.14E+02	3.68E-01	2.94E-01	2.77E-01	3.15E-01	2.97E-01
Y-91m	5.46E+01	1.03E+02	5.94E+01	4.77E+01	4.45E+01	5.51E+01	4.87E+01
Y-92	2.42E+01	1.60E+03	2.68E+01	2.14E+01	2.01E+01	2.34E+01	2.17E+01
Y-93	8.55E+00	1.28E+03	9.31E+00	7.52E+00	7.08E+00	8.45E+00	7.64E+00
Zr-95+D	2.45E+02	3.36E+02	2.71E+02	2.16E+02	2.02E+02	2.40E+02	2.19E+02
Zr-97+D	1.61E+02	1.41E+03	1.78E+02	1.41E+02	1.32E+02	1.59E+02	1.44E+02
Nb-95	7.76E+01	1.06E+02	8.61E+01	6.84E+01	6.40E+01	7.60E+01	6.95E+01
Nb-96	2.45E+02	4.65E+02	2.71E+02	2.16E+02	2.02E+02	2.40E+02	2.19E+02
Mo-99+D	1.60E+01	3.55E+02	1.77E+01	1.39E+01	1.30E+01	1.62E+01	1.40E+01
Tc-99m	1.48E+01	1.82E+01	1.64E+01	1.19E+01	1.08E+01	1.87E+01	1.09E+01
Ru-103+D	5.01E+01	7.36E+01	5.42E+01	4.35E+01	4.05E+01	5.11E+01	4.45E+01
Ru-105+D	8.14E+01	4.53E+02	8.94E+01	7.11E+01	6.63E+01	8.20E+01	7.24E+01
Ru-106+D	2.10E+01	1.56E+03	2.29E+01	1.84E+01	1.72E+01	2.11E+01	1.88E+01
Sb-125+D	4.64E+01	8.09E+01	5.00E+01	3.91E+01	3.62E+01	4.68E+01	3.92E+01
Sb-129+D	1.46E+02	1.04E+03	1.62E+02	1.29E+02	1.21E+02	1.44E+02	1.32E+02
Te-131	4.36E+01	7.68E+02	4.81E+01	3.74E+01	3.47E+01	4.64E+01	3.72E+01
Te-131m+D	1.51E+02	4.19E+02	1.67E+02	1.32E+02	1.24E+02	1.49E+02	1.33E+02
Te-132+D	2.61E+02	8.14E+02	2.87E+02	2.27E+02	2.13E+02	2.59E+02	2.31E+02
Te-133m+D	2.34E+02	1.16E+03	2.60E+02	2.06E+02	1.94E+02	2.31E+02	2.09E+02
Te-134	9.08E+01	1.26E+02	9.98E+01	7.84E+01	7.29E+01	9.41E+01	7.90E+01
I-129	2.20E+00	5.28E+00	2.17E+00	1.07E+00	8.43E-01	1.87E+00	4.02E-01
I-131	4.04E+01	1.06E+02	4.38E+01	3.47E+01	3.22E+01	4.28E+01	3.57E+01
I-132	2.29E+02	7.58E+02	2.53E+02	2.01E+02	1.89E+02	2.24E+02	2.05E+02
I-133	6.16E+01	4.38E+02	6.72E+01	5.39E+01	5.04E+01	6.20E+01	5.51E+01
I-134	2.58E+02	9.51E+02	2.86E+02	2.28E+02	2.14E+02	2.51E+02	2.31E+02
I-135	1.45E+02	4.95E+02	1.59E+02	1.29E+02	1.21E+02	1.38E+02	1.30E+02
Xe-129m	4.76E+00	1.04E+01	4.79E+00	2.64E+00	2.18E+00	4.35E+00	1.52E+00
Xe-131m	1.83E+00	4.11E+00	1.82E+00	9.85E-01	8.00E-01	1.65E+00	5.41E-01
Xe-133	5.08E+00	7.97E+00	5.68E+00	3.56E+00	3.15E+00	6.30E+00	2.22E+00
Xe-133m	4.15E+00	7.13E+00	4.35E+00	2.92E+00	2.58E+00	4.27E+00	2.51E+00
Xe-135	2.71E+01	2.44E+02	2.94E+01	2.27E+01	2.10E+01	3.03E+01	2.32E+01
Xe-135m	4.43E+01	1.64E+02	4.81E+01	3.85E+01	3.59E+01	4.50E+01	3.93E+01
Xe-137	1.89E+01	1.95E+03	2.05E+01	1.65E+01	1.54E+01	1.92E+01	1.69E+01
Xe-138	1.05E+02	7.34E+02	1.14E+02	9.33E+01	8.81E+01	1.03E+02	9.46E+01

Table C-2. (continued)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Xe-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.58E+02	2.76E+02	1.74E+02	1.39E+02	1.30E+02	1.56E+02	1.41E+02
Cs-136	2.14E+02	2.94E+02	2.38E+02	1.88E+02	1.76E+02	2.12E+02	1.90E+02
Cs-137+D	5.78E+01	1.84E+02	6.36E+01	5.07E+01	4.73E+01	5.73E+01	5.17E+01
Cs-138	2.15E+02	1.60E+03	2.34E+02	1.91E+02	1.81E+02	2.05E+02	1.93E+02
Cs-139	2.73E+01	1.83E+03	2.97E+01	2.43E+01	2.31E+01	2.58E+01	2.46E+01
Ba-139	3.88E+00	9.56E+02	4.29E+00	3.19E+00	2.92E+00	4.60E+00	3.04E+00
Ba-140+D	2.65E+02	1.12E+03	2.89E+02	2.34E+02	2.22E+02	2.56E+02	2.37E+02
Ba-141	8.94E+01	1.00E+03	9.80E+01	7.76E+01	7.25E+01	9.21E+01	7.87E+01
Ba-142	8.82E+01	5.22E+02	9.78E+01	7.73E+01	7.24E+01	8.74E+01	7.78E+01
La-141	3.79E+00	1.01E+03	4.17E+00	3.37E+00	3.19E+00	3.59E+00	3.40E+00
La-142	2.44E+02	1.20E+03	2.61E+02	2.17E+02	2.07E+02	2.30E+02	2.22E+02
Ce-141	8.78E+00	2.75E+01	9.80E+00	7.02E+00	6.37E+00	1.10E+01	6.31E+00
Ce-143	2.85E+01	3.92E+02	3.12E+01	2.39E+01	2.21E+01	3.03E+01	2.35E+01
Ce-144+D	5.10E+00	1.29E+03	5.60E+00	4.29E+00	3.98E+00	5.52E+00	4.04E+00
Pr-143	9.06E-07	2.27E+02	1.00E-06	7.97E-07	7.46E-07	8.90E-07	8.11E-07
Pr-144	2.98E+00	1.31E+03	3.22E+00	2.65E+00	2.51E+00	2.84E+00	2.69E+00
Pm-147	4.10E-04	4.96E-04	4.63E-04	3.30E-04	2.98E-04	5.37E-04	2.85E-04
Eu-154	1.21E+02	2.97E+02	1.34E+02	1.06E+02	9.97E+01	1.19E+02	1.06E+02
Eu-155	7.15E+00	9.39E+00	8.22E+00	5.47E+00	4.93E+00	9.53E+00	3.90E+00
Ta-182	1.23E+02	1.78E+02	1.37E+02	1.08E+02	1.01E+02	1.22E+02	1.06E+02
Ir-192	8.70E+01	1.73E+02	9.43E+01	7.46E+01	6.92E+01	9.27E+01	7.64E+01
Hg-203	2.50E+01	3.22E+01	2.72E+01	2.10E+01	1.94E+01	2.81E+01	2.12E+01
Th-232	9.25E+01	4.34E+02	1.03E+02	8.11E+01	7.60E+01	9.10E+01	8.19E+01
Np-237+D	2.68E+01	4.03E+01	2.87E+01	2.18E+01	1.99E+01	3.03E+01	2.08E+01
U-234	8.07E-02	6.36E-01	2.31E-02	1.74E-02	9.34E-03	2.95E-02	9.15E-03
U-235+D	1.90E+01	2.91E+01	2.05E+01	1.50E+01	1.36E+01	2.26E+01	1.41E+01
U-238+D	2.20E+00	8.76E+02	2.38E+00	1.75E+00	1.60E+00	2.44E+00	1.53E+00
Pu-238	8.58E-02	7.17E-01	1.51E-02	1.22E-02	2.63E-03	2.05E-02	4.51E-03
Pu-239	3.78E-02	2.80E-01	1.16E-02	8.94E-03	4.85E-03	1.47E-02	5.58E-03
Pu-240	8.20E-02	6.83E-01	1.49E-02	1.19E-02	2.77E-03	2.00E-02	4.43E-03
Pu-239/240	3.78E-02	2.80E-01	1.16E-02	8.94E-03	4.85E-03	1.47E-02	5.58E-03

a. Dose conversion factors are from DOE (1988a) (bone surface = skeleton).

Table C-3. Dose conversion factors for dose equivalent from inhalation (adults) (rem per μCi intake)[a]

Radio-nuclide	Aerosol Class	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	HTO	6.30E-05	6.30E-05	6.30E-05	6.30E-05	6.30E-05	0.00E+00	6.30E-05
C-14	ORG	2.10E-03	0.00E+00	2.10E-03	2.10E-03	0.00E+00	2.10E-03	2.10E-03
N-13	No DCFs[b]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	D	9.50E-04	0.00E+00	0.00E+00	4.40E-03	0.00E+00	0.00E+00	7.80E-04
Ar-41	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sc-46	Y	2.00E-02	0.00E+00	0.00E+00	1.70E-01	0.00E+00	0.00E+00	0.00E+00
Cr-51	Y	2.60E-04	0.00E+00	0.00E+00	2.00E-03	0.00E+00	0.00E+00	0.00E+00
Mn-54	W	6.40E-03	0.00E+00	0.00E+00	2.50E-02	9.30E-03	0.00E+00	4.10E-03
Mn-56	D	3.30E-04	0.00E+00	0.00E+00	1.60E-03	0.00E+00	0.00E+00	0.00E+00
Co-58	Y	7.10E-03	0.00E+00	0.00E+00	5.90E-02	0.00E+00	0.00E+00	0.00E+00
Co-60	Y	1.50E-01	0.00E+00	0.00E+00	1.30E+00	0.00E+00	0.00E+00	0.00E+00
As-76	W	3.40E-03	0.00E+00	0.00E+00	1.90E-02	0.00E+00	0.00E+00	0.00E+00
Br-82	W	1.30E-03	0.00E+00	0.00E+00	6.30E-03	0.00E+00	1.40E-03	8.10E-04
Br-84	D	8.70E-05	0.00E+00	0.00E+00	5.90E-04	0.00E+00	0.00E+00	0.00E+00
Kr-85	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-90	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	D	3.70E-05	0.00E+00	0.00E+00	2.50E-04	0.00E+00	0.00E+00	0.00E+00
Rb-90	No DCFs	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Y	3.70E-02	0.00E+00	0.00E+00	3.10E-01	0.00E+00	0.00E+00	0.00E+00
Sr-90	Y	1.30E+00	0.00E+00	0.00E+00	1.10E+01	0.00E+00	0.00E+00	0.00E+00
Sr-91	Y	1.40E-03	0.00E+00	0.00E+00	7.80E-03	0.00E+00	0.00E+00	0.00E+00
Sr-92	Y	7.70E-04	0.00E+00	0.00E+00	4.10E-03	0.00E+00	0.00E+00	0.00E+00
Y-91	Y	4.40E-02	0.00E+00	0.00E+00	3.70E-01	0.00E+00	0.00E+00	0.00E+00
Y-91m	Y	3.10E-05	0.00E+00	0.00E+00	2.60E-04	0.00E+00	0.00E+00	0.00E+00
Y-92	Y	6.20E-04	0.00E+00	0.00E+00	4.40E-03	0.00E+00	0.00E+00	0.00E+00
Y-93	Y	2.10E-03	0.00E+00	0.00E+00	9.30E-03	0.00E+00	0.00E+00	0.00E+00
Zr-95	D	1.90E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.70E-01	4.80E-02
Zr-97	Y	4.00E-03	0.00E+00	0.00E+00	1.50E-02	0.00E+00	0.00E+00	0.00E+00
Nb-95	Y	4.50E-03	0.00E+00	0.00E+00	3.10E-02	0.00E+00	0.00E+00	0.00E+00
Nb-96	Y	2.00E-03	0.00E+00	0.00E+00	5.90E-03	0.00E+00	0.00E+00	0.00E+00
Mo-99	Y	3.60E-03	0.00E+00	0.00E+00	1.60E-02	0.00E+00	0.00E+00	0.00E+00
Tc-99m	D	3.20E-05	0.00E+00	1.80E-04	8.50E-05	0.00E+00	0.00E+00	1.30E-05
Ru-103	Y	7.80E-03	0.00E+00	0.00E+00	5.90E-02	0.00E+00	0.00E+00	0.00E+00
Ru-105	Y	4.10E-04	0.00E+00	0.00E+00	2.10E-03	0.00E+00	0.00E+00	0.00E+00
Ru-106	Y	4.40E-01	0.00E+00	0.00E+00	3.70E+00	0.00E+00	0.00E+00	0.00E+00
Sb-125	W	9.80E-03	0.00E+00	0.00E+00	8.10E-02	0.00E+00	0.00E+00	0.00E+00
Sb-129	W	5.70E-04	0.00E+00	0.00E+00	3.30E-03	0.00E+00	0.00E+00	0.00E+00
Te-131	W	9.90E-05	0.00E+00	2.10E-03	3.00E-04	0.00E+00	0.00E+00	0.00E+00
Te-131m	W	5.50E-03	0.00E+00	1.20E-01	8.10E-03	0.00E+00	0.00E+00	0.00E+00
Te-132	W	7.70E-03	0.00E+00	2.30E-01	6.30E-03	0.00E+00	0.00E+00	0.00E+00
Te-133m	W	3.80E-04	0.00E+00	9.60E-03	7.80E-04	0.00E+00	0.00E+00	0.00E+00
Te-134	D	1.00E-04	0.00E+00	2.00E-03	2.20E-04	0.00E+00	0.00E+00	0.00E+00
I-129	D	1.80E-01	0.00E+00	5.90E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	D	3.20E-02	0.00E+00	1.10E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	D	3.30E-04	0.00E+00	6.30E-03	1.00E-03	0.00E+00	0.00E+00	0.00E+00
I-133	D	5.40E-03	0.00E+00	1.80E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	D	1.10E-04	0.00E+00	1.10E-03	5.20E-04	0.00E+00	0.00E+00	0.00E+00
I-135	D	1.10E-03	0.00E+00	3.10E-02	1.60E-03	0.00E+00	0.00E+00	0.00E+00
Xe-129m	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-137	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-3. (continued)

Radio-nuclide	Aerosol Class	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Xe-139	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	D	4.70E-02	0.00E+00	4.10E-02	4.40E-02	0.00E+00	4.10E-02	4.40E-02
Cs-136	D	7.50E-03	0.00E+00	6.30E-03	8.50E-03	0.00E+00	6.30E-03	7.00E-03
Cs-137	D	3.20E-02	0.00E+00	2.90E-02	3.30E-02	0.00E+00	2.90E-02	3.10E-02
Cs-138	D	8.80E-05	0.00E+00	0.00E+00	5.90E-04	0.00E+00	0.00E+00	0.00E+00
Cs-139	No DCFs	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	D	1.60E-04	0.00E+00	0.00E+00	9.30E-04	0.00E+00	0.00E+00	0.00E+00
Ba-140	D	3.60E-03	0.00E+00	0.00E+00	6.30E-03	0.00E+00	8.90E-03	4.80E-03
Ba-141	D	7.40E-05	0.00E+00	0.00E+00	4.40E-04	0.00E+00	0.00E+00	0.00E+00
Ba-142	D	3.60E-05	0.00E+00	0.00E+00	2.00E-04	0.00E+00	0.00E+00	0.00E+00
La-141	D	5.40E-04	0.00E+00	0.00E+00	2.40E-03	7.80E-04	0.00E+00	0.00E+00
La-142	D	2.20E-04	0.00E+00	0.00E+00	1.10E-03	0.00E+00	0.00E+00	0.00E+00
Ce-141	Y	8.50E-03	0.00E+00	0.00E+00	6.30E-02	0.00E+00	0.00E+00	0.00E+00
Ce-143	Y	3.20E-03	0.00E+00	0.00E+00	1.40E-02	0.00E+00	0.00E+00	0.00E+00
Ce-144	Y	3.50E-01	0.00E+00	0.00E+00	2.90E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	Y	7.30E-03	0.00E+00	0.00E+00	4.80E-02	0.00E+00	0.00E+00	0.00E+00
Pr-144	Y	4.20E-05	0.00E+00	0.00E+00	3.50E-04	0.00E+00	0.00E+00	0.00E+00
Pm-147	Y	3.40E-02	0.00E+00	0.00E+00	2.80E-01	0.00E+00	0.00E+00	0.00E+00
Eu-154	W	2.60E-01	0.00E+00	0.00E+00	2.90E-01	1.60E+00	1.90E+00	4.10E-01
Eu-155	W	3.90E-02	0.00E+00	0.00E+00	4.40E-02	1.80E-01	5.60E-01	5.20E-02
Ta-182	Y	3.70E-02	0.00E+00	0.00E+00	3.10E-01	0.00E+00	0.00E+00	0.00E+00
Ir-192	Y	2.30E-02	0.00E+00	0.00E+00	1.90E-01	0.00E+00	0.00E+00	0.00E+00
Hg-203	D (ORG)	6.50E-03	0.00E+00	0.00E+00	4.10E-03	0.00E+00	0.00E+00	4.10E-03
Th-232	W	1.60E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.10E+04	3.30E+03
Np-237	W	4.90E+02	0.00E+00	0.00E+00	0.00E+00	1.90E+03	8.90E+03	7.00E+02
U-234	Y	1.30E+02	0.00E+00	0.00E+00	1.10E+03	0.00E+00	0.00E+00	0.00E+00
U-235	Y	1.20E+02	0.00E+00	0.00E+00	1.00E+03	0.00E+00	0.00E+00	0.00E+00
U-238	Y	1.20E+02	0.00E+00	0.00E+00	1.00E+03	0.00E+00	0.00E+00	0.00E+00
Pu-238	W	4.60E+02	0.00E+00	0.00E+00	0.00E+00	1.80E+03	8.10E+03	6.70E+02
Pu-239	W	5.10E+02	0.00E+00	0.00E+00	0.00E+00	2.00E+03	9.30E+03	7.40E+02
Pu-240	W	5.10E+02	0.00E+00	0.00E+00	0.00E+00	2.00E+03	9.30E+03	7.40E+02
Pu-239/240	W	5.10E+02	0.00E+00	0.00E+00	0.00E+00	2.00E+03	9.30E+03	7.40E+02

a. Dose conversion factors are from DOE (1988b). The H-3 dose conversion factor for all organs except the bone surface is the dose conversion factor for soft tissue.

b. DCF = dose conversion factor.

Table C-4. Dose conversion factors for dose equivalent from ingestion (adults) (rem per μCi intake)[a]

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	6.30E-05	6.30E-05	6.30E-05	6.30E-05	6.30E-05	0.00E+00	6.30E-05
C-14	2.10E-03	0.00E+00	2.10E-03	2.10E-03	0.00E+00	2.10E-03	2.10E-03
N-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	1.40E-03	0.00E+00	0.00E+00	9.60E-04	0.00E+00	1.70E-03	1.40E-03
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sc-46	5.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.30E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.40E-05
Mn-54	2.70E-03	0.00E+00	0.00E+00	8.50E-04	3.70E-03	0.00E+00	1.80E-03
Mn-56	9.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	3.50E-03	0.00E+00	0.00E+00	1.50E-03	3.70E-03	0.00E+00	2.00E-03
Co-60	2.60E-02	0.00E+00	0.00E+00	1.90E-02	4.80E-02	0.00E+00	2.00E-02
As-76	4.80E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-82	1.70E-03	0.00E+00	0.00E+00	1.40E-03	0.00E+00	1.40E-03	1.50E-03
Br-84	1.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	8.00E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	8.70E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.30E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.60E+00	7.00E-01
Sr-91	3.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.90E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	8.90E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	3.90E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	1.90E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	4.50E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	3.40E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-97	8.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	2.20E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	4.40E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	4.40E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	6.00E-05	0.00E+00	3.10E-04	0.00E+00	0.00E+00	0.00E+00	2.30E-05
Ru-103	2.70E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-105	1.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	2.10E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-125	2.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-129	1.70E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	2.00E-04	0.00E+00	3.30E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.50E-02	0.00E+00	1.40E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	7.40E-03	0.00E+00	2.20E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	7.60E-04	0.00E+00	1.60E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	2.10E-04	0.00E+00	3.30E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129	2.80E-01	0.00E+00	9.30E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	5.30E-02	0.00E+00	1.80E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	5.70E-04	0.00E+00	1.40E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	1.00E-02	0.00E+00	3.40E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.90E-04	0.00E+00	2.30E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.00E-03	0.00E+00	6.70E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-137	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-4. (continued)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Xe-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	7.40E-02	0.00E+00	6.70E-02	6.70E-02	0.00E+00	6.30E-02	7.00E-02
Cs-136	1.10E-02	0.00E+00	1.00E-02	9.60E-03	0.00E+00	1.00E-02	1.10E-02
Cs-137	5.00E-02	0.00E+00	4.80E-02	4.80E-02	0.00E+00	4.80E-02	4.80E-02
Cs-138	1.60E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.90E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	8.40E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	2.00E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	1.00E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	1.40E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	6.30E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	2.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	4.20E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	2.00E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	4.50E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.10E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	9.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	9.10E-03	0.00E+00	0.00E+00	0.00E+00	1.40E-02	1.70E-02	4.10E-03
Eu-155	1.30E-03	0.00E+00	0.00E+00	0.00E+00	1.50E-03	4.80E-03	0.00E+00
Ta-182	6.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ir-192	5.30E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Hg-203	1.00E-02	0.00E+00	0.00E+00	4.40E-03	0.00E+00	0.00E+00	6.30E-03
Th-232	2.80E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.00E+01	5.60E+00
Np-237	3.90E+00	0.00E+00	0.00E+00	0.00E+00	1.50E+01	7.00E+01	5.60E+00
U-234	2.60E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.10E+00	2.70E-01
U-235	2.50E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.70E+00	2.50E-01
U-238	2.30E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.70E+00	2.50E-01
Pu-238	3.80E+00	0.00E+00	0.00E+00	0.00E+00	1.50E+01	6.70E+01	5.60E+00
Pu-239	4.30E+00	0.00E+00	0.00E+00	0.00E+00	1.60E+01	7.80E+01	5.90E+00
Pu-240	4.30E+00	0.00E+00	0.00E+00	0.00E+00	1.60E+01	7.80E+01	5.90E+00
Pu-239/240	4.30E+00	0.00E+00	0.00E+00	0.00E+00	1.60E+01	7.80E+01	5.90E+00

a. Dose conversion factors are from DOE (1988a). The H-3 dose conversion factor for all organs except the bone surface is the dose conversion factor for soft tissue.

Table C-5. Dose conversion factors for dose equivalent from inhalation (infants and children) (rem per μCi intake)[a]

Radio-nuclide	Aerosol Class	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Infants								
H-3	HTO	2.04E-04	2.04E-04	2.04E-04	2.04E-04	2.04E-04	2.04E-04	2.04E-04
C-14	ORG	4.81E-03	4.81E-03	4.81E-03	4.81E-03	4.81E-03	4.81E-03	4.81E-03
Sr-90	D	4.44E-01	3.70E-02	3.70E-02	8.51E-02	3.70E-02	3.52E+00	2.52E+00
Zr-95	Y	1.85E-01	1.41E-02	2.78E-02	1.18E+00	4.81E-02	9.25E-02	3.63E-02
Nb-95	Y	4.44E-02	4.81E-03	8.88E-03	2.41E-01	1.70E-02	1.55E-02	8.88E-03
Ru-103	Y	8.14E-02	4.81E-03	8.14E-03	5.18E-01	1.30E-02	8.88E-03	7.40E-03
Ru-106	Y	3.63E+00	5.55E-02	6.29E-02	2.89E+01	7.40E-02	7.40E-02	8.80E-02
I-129	D	2.59E-01	5.18E-04	8.51E+00	9.62E-03	5.18E-04	1.41E-03	8.14E-04
I-131	D	2.55E-01	1.11E-03	8.51E+00	2.59E-02	1.22E-03	1.63E-03	1.67E-03
I-132	D	3.70E-03	2.92E-04	6.66E-02	1.04E-02	5.18E-04	4.44E-04	4.44E-04
Cs-134	D	6.29E-02	4.07E-02	5.92E-02	7.77E-02	5.92E-02	5.92E-02	6.29E-02
Cs-137	D	4.81E-02	3.66E-02	4.44E-02	7.40E-02	4.44E-02	4.81E-02	6.29E-02
Ce-144	Y	2.92E+00	9.62E-03	1.15E-02	2.29E+01	2.04E-01	3.55E-01	4.07E-01
Pu-238	W	6.66E+02	2.70E+01	2.70E+01	7.03E+02	3.03E+03	7.40E+03	9.62E+02
Pu-239	W	7.03E+02	2.66E+01	2.66E+01	6.66E+02	3.33E+03	8.14E+03	9.99E+02
Pu-240	W	7.03E+02	2.66E+01	2.66E+01	6.66E+02	3.33E+03	8.14E+03	9.99E+02
Pu-239/240	W	7.03E+02	2.66E+01	2.66E+01	6.66E+02	3.33E+03	8.14E+03	9.99E+02
Children								
H-3	HTO	7.03E-05	7.03E-05	7.03E-05	7.03E-05	7.03E-05	7.03E-05	7.03E-05
C-14	ORG	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03
Sr-90	D	2.26E-01	1.37E-02	1.37E-02	2.29E-02	1.37E-02	3.07E+00	9.99E-01
Zr-95	Y	4.81E-02	4.07E-03	8.14E-03	3.07E-01	1.52E-02	2.18E-02	1.04E-02
Nb-95	Y	1.22E-02	1.33E-03	2.52E-03	6.29E-02	5.18E-03	3.26E-03	2.92E-03
Ru-103	Y	1.92E-02	1.04E-03	2.00E-03	1.22E-01	3.59E-03	1.92E-03	2.15E-03
Ru-106	Y	9.99E-01	9.62E-03	1.22E-02	8.14E+00	1.59E-02	1.30E-02	1.37E-02
I-129	D	2.66E-01	3.55E-04	8.88E+00	2.26E-03	3.15E-04	1.04E-03	4.07E-04
I-131	D	7.40E-02	2.85E-04	2.41E+00	5.18E-03	2.81E-04	5.18E-04	4.07E-04
I-132	D	8.14E-04	6.66E-05	1.41E-02	2.15E-03	1.30E-04	9.99E-05	9.99E-05
Cs-134	D	3.40E-02	2.15E-02	3.29E-02	3.55E-02	3.33E-02	3.40E-02	3.29E-02
Cs-137	D	2.41E-02	1.81E-02	2.26E-02	2.81E-02	2.26E-02	2.44E-02	2.48E-02
Ce-144	Y	7.77E-01	2.07E-03	2.63E-03	6.29E+00	1.18E-01	5.55E-02	4.44E-02
Pu-238	W	4.07E+02	8.14E+00	8.14E+00	1.55E+02	1.78E+03	6.66E+03	4.07E+02
Pu-239	W	4.44E+02	8.51E+00	8.51E+00	1.48E+02	2.04E+03	7.77E+03	4.44E+02
Pu-240	W	4.44E+02	8.51E+00	8.51E+00	1.48E+02	2.04E+03	7.77E+03	4.44E+02
Pu-239/240	W	4.44E+02	8.51E+00	8.51E+00	1.48E+02	2.04E+03	7.77E+03	4.44E+02

a. Source: ICRP (1989).

Table C-6. Dose conversion factors for dose equivalent from ingestion (infants and children) (rem per μCi intake)[a]

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Infants							
H-3	2.04E-04	2.03E-04	2.03E-04	2.03E-04	2.03E-04	2.03E-04	2.03E-04
C-14	4.81E-03	4.81E-03	4.81E-03	4.81E-03	4.81E-03	4.81E-03	4.81E-03
Sr-90	4.81E-01	4.07E-02	4.07E-02	4.07E-02	4.07E-02	3.70E+00	2.63E+00
Zr-95	3.70E-02	2.44E-03	1.41E-03	2.44E-03	4.07E-03	8.51E-02	2.33E-02
Nb-95	1.92E-02	1.41E-03	6.29E-04	1.33E-03	4.81E-03	1.04E-02	4.44E-03
Ru-103	2.85E-02	3.63E-03	4.07E-03	4.07E-03	5.55E-03	5.18E-03	5.92E-03
Ru-106	3.29E-01	8.14E-02	8.51E-02	8.51E-02	8.51E-02	9.99E-02	1.33E-01
I-129	4.07E-01	8.14E-04	1.37E+01	1.55E-03	7.77E-04	2.11E-03	1.26E-03
I-131	4.07E-01	1.74E-03	1.37E+01	3.29E-03	1.70E-03	2.44E-03	2.59E-03
I-132	6.66E-03	5.55E-04	1.52E-01	1.11E-03	8.88E-04	7.77E-04	8.88E-04
Cs-134	9.25E-02	6.29E-02	9.25E-02	8.14E-02	9.25E-02	9.62E-02	9.99E-02
Cs-137	7.40E-02	5.92E-02	7.03E-02	6.66E-02	7.03E-02	7.77E-02	9.99E-02
Ce-144	2.96E-01	5.92E-03	5.92E-03	5.92E-03	8.88E-02	2.55E-01	3.26E-01
Pu-238	4.81E+01	2.22E+00	2.22E+00	2.26E+00	2.41E+02	5.92E+02	8.14E+01
Pu-239	5.18E+01	2.22E+00	2.22E+00	2.22E+00	2.63E+02	6.66E+02	8.14E+01
Pu-240	5.18E+01	2.22E+00	2.22E+00	2.22E+00	2.63E+02	6.66E+02	8.14E+01
Pu-239/240	5.18E+01	2.22E+00	2.22E+00	2.22E+00	2.63E+02	6.66E+02	8.14E+01
Children							
H-3	7.03E-05	7.03E-05	7.03E-05	7.03E-05	7.03E-05	7.03E-05	7.03E-05
C-14	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03
Sr-90	1.59E-01	9.25E-03	9.25E-03	9.25E-03	9.25E-03	2.04E+00	6.66E-01
Zr-95	8.14E-03	4.44E-04	2.44E-04	4.07E-04	8.51E-04	1.44E-02	3.18E-03
Nb-95	4.81E-03	2.92E-04	9.62E-05	2.37E-04	1.07E-03	1.37E-03	1.07E-03
Ru-103	6.29E-03	4.81E-04	4.81E-04	5.18E-04	8.51E-04	7.77E-04	1.04E-03
Ru-106	5.92E-02	1.04E-02	1.04E-02	1.04E-02	1.04E-02	1.18E-02	1.30E-02
I-129	4.07E-01	5.55E-04	1.41E+01	8.51E-04	4.81E-04	1.63E-03	6.66E-04
I-131	1.18E-01	4.44E-04	4.07E+00	8.88E-04	3.63E-04	7.77E-04	5.92E-04
I-132	1.41E-03	1.22E-04	3.15E-02	2.48E-04	2.07E-04	1.74E-04	1.78E-04
Cs-134	5.18E-02	3.40E-02	5.18E-02	4.81E-02	5.18E-02	5.55E-02	5.18E-02
Cs-137	3.63E-02	2.89E-02	3.59E-02	3.40E-02	3.59E-02	4.07E-02	4.07E-02
Ce-144	4.81E-02	1.67E-04	1.52E-04	1.67E-04	1.07E-02	5.18E-03	4.07E-03
Pu-238	3.26E+00	7.03E-02	7.03E-02	7.03E-02	1.48E+01	5.55E+01	3.44E+00
Pu-239	3.70E+00	7.03E-02	7.03E-02	7.40E-02	1.67E+01	6.29E+01	3.70E+00
Pu-240	3.70E+00	7.03E-02	7.03E-02	7.40E-02	1.67E+01	6.29E+01	3.70E+00
Pu-239/240	3.70E+00	7.03E-02	7.03E-02	7.40E-02	1.67E+01	6.29E+01	3.70E+00

a. Source: ICRP (1989).

integrated concentration on the ground surface, which is obtained by multiplying the total integrated air concentration by a deposition velocity. Concentration of radioactivity in the air at the receptor point is, therefore, the starting point for calculating all four dose modes of exposure.

The methods for estimating total integrated air concentrations from annual operational releases and from short-term episodic releases differ slightly because of differences in the way atmospheric dispersion is treated for the two cases. For operational releases, an annualized mesoscale dispersion factor (MDF) is developed using the methods described in Appendix B. This MDF is multiplied by the annual release rate and corrected for radioactive decay during atmospheric transit to the INEL Site boundary to calculate air concentration according to the following equation:

$$TIC = \frac{Q_o}{T} MDF_o \, e^{-\lambda_r \frac{d}{v}} \qquad \text{(C-1)}$$

where

TIC	=	total integrated air concentration (Ci-h/m^3)
Q_o	=	annual release of radioactivity (Ci)
T	=	period of release, 1 year (h)
MDF_o	=	annual average mesoscale dispersion factor (h^2/m^3)
λ_r	=	radiological decay constant (h^{-1})
d	=	distance from release point to receptor (km)
v	=	average wind speed toward receptor (km/h).

The average air concentration over the year at the receptor location is equal to the total integrated air concentration divided by the exposure period. For

operational releases, the exposure period equals the period of release, 1 year. From opertaional releases, the average air concentration for each radionuclide is given by

$$ACC = \frac{TIC}{T} \qquad \text{(C-2)}$$

where

AAC	=	average air concentration (Ci/m^3)
TIC	=	total integrated air concentration ($Ci\text{-}h/m^3$)
T	=	period of release, 1 year (h).

For the episodic releases, the MDF is calculated for the specific conditions during each release according to the method described in Appendix B. Each episodic MDF is based on the period of time needed to model the trajectory of the released puff. The transit time from release point to the INEL Site boundary is also calculated for each release. While developing the source term, the amount of radioactivity released in each episode is corrected for radioactive decay and daughter ingrowth during atmospheric transit to the INEL Site boundary using the methods described in Appendix A. This decay-corrected source term is used to calculate the total integrated air concentration for episodic releases according to the following equation:

$$TIC = Q_e \, MDF_e \qquad \text{(C-3)}$$

where

TIC	=	total integrated air concentration ($Ci\text{-}h/m^3$)
Q_e	=	release of radioactivity during episode corrected for radioactive decay to the site boundary (Ci)

MDF_e = episodic mesoscale dispersion factor (h/m^3).

The average air concentration over the entire exposure period for each episode at the receptor location is equal to the total integrated air concentration divided by the modeled time period. For operational releases, the exposure period equals the period of release, 1 year. However, for episodic releases, the exposure period is longer than the period over which the release occurred to allow for the transport of the radioactivity to the receptor location and for the plume to completely pass the receptor location. For episodic releases, the average air concentration for each radionuclide is given by

$$AAC = \frac{TIC}{t_e} \tag{C-4}$$

where

AAC = average air concentration (Ci/m^3)

TIC = total integrated air concentration ($Ci\text{-}h/m^3$)

t_e = modeled time period (h) = exposure period (h).

The exposure period is the time over which the MDF is calculated; therefore, it is the period when exposures via air immersion, inhalation, and deposition on ground surfaces occur.

C.2.2.3 Air Immersion Dose

External doses because of immersion in contaminated air are calculated using the total integrated concentration given by Equation (C-1) for operational releases or Equation (C-3) for episodic releases and appropriate dose conversion factors from Table C-1 (DOE, 1988b). No credit is taken for shielding from buildings during the portion of the year spent indoors. Air immersion dose is given by

$$D_a = TIC\ DCF_a\ (1.00 \times 10^6)\ (1.14 \times 10^{-4}) \qquad \text{(C-5)}$$

where

D_a	=	air immersion dose (mrem)
TIC	=	total integrated air concentration (Ci-h/m^3)
DCF_a	=	dose conversion factor (mrem/y per μCi/m^3)
1.00×10^6	=	unit conversion (Ci to μCi)
1.14×10^{-4}	=	unit conversion (h to y).

C.2.2.4 Inhalation Dose

Internal doses because of breathing contaminated air are calculated using the total integrated air concentration given by Equation (C-1) for operational releases or Equation (C-3) for episodic releases. Appropriate dose conversion factors for inhalation are taken from Table C-3, with the exception of age-specific values taken from Table C-5 for inhalation of some radionuclides by children and infants. The assumed breathing rates for inhalation doses are 8400 m^3/yr for adults, 5475 m^3/yr for children, and 1387 m^3/yr for infants (ICRP, 1975). For each radionuclide and age group, inhalation dose is given by

$$D_i = TIC\ BR_j\ DCF_{ij}\ (1.00 \times 10^6)\ (1.00 \times 10^3) \qquad \text{(C-6)}$$

where

D_i	=	inhalation dose (mrem)
TIC	=	total integrated air concentration (Ci-h/m^3)

BR_j = breathing rate for age group j (m^3/h)

DCF_{ij} = inhalation dose conversion factor for age group j (rem/μCi)

1.00×10^6 = unit conversion (Ci to μCi)

1.00×10^3 = unit conversion (rem to mrem).

C.2.2.5 Ground Surface Dose

External doses because of contaminated ground surfaces are calculated using integrated ground deposition based on the average air concentration, given by Equation (C-2) for operational releases or Equation (C-4) for episodic releases and the deposition velocity. Deposition velocities for particulates are assumed to be 0.18 cm/s. For the elemental radioiodines, a deposition velocity of 1.8 cm/s is used. The deposition velocity for organic radioiodines is 0.018 cm/s (DOE, 1987). These deposition velocities are used to calculate total deposition on both vegetation and the ground surface. They correspond to deposition velocities for pasture grass of 0.1, 1.0, and 0.01 cm/s, respectively. Dose conversion factors are taken from Table C-2.

Dose from contaminated ground surfaces is evaluated not only for the exposure period of the release but also for an additional time period during which radioactive materials may persist on the ground. The equation for ground surface dose includes the results of integration over the period of deposition and the following dose commitment period to account for radioactive decay. Dose from contaminated ground surfaces is given by

$$D_S = AAC\ V_d\ SF\ DCF_s\ \frac{1}{\lambda_r}\left(t - \frac{e^{-\lambda_r T}}{\lambda_r}\ (1 - e^{\lambda_r t})\right) \times (3.60 \times 10^3)\ (1.00 \times 10^6)\ (1.14 \times 10^{-4}) \qquad \text{(C-7)}$$

where

D_s	=	ground surface dose (mrem)
AAC	=	average air concentration (Ci/m^3)
V_d	=	deposition velocity (m/s)
SF	=	residential shielding factor (unitless)
DCF_s	=	dose conversion factor (mrem/yr per $\mu Ci/m^2$)
λ_r	=	radiological decay constant (h^{-1})
t	=	exposure period (h)
T	=	dose commitment period for surface deposition (h)
3.60×10^3	=	unit conversion (h to s)
1.00×10^6	=	unit conversion (Ci to μCi)
1.14×10^{-4}	=	unit conversion (h to yr).

For this evaluation, the dose commitment period for which exposure to ground surface contamination was calculated is 50 years. The radioactivity is assumed to remain on the surface of the soil, and no credit is taken for loss of radioactivity from soil by mechanisms other than radioactive decay (e.g., weathering or erosion) during this period. The residential shielding factor of 0.7 is used, and it is consistent with the standard practice for maximum individual dose calculation (DOE, 1988a). The total dose for the 50-year period is assigned to the year of the release, even though for some radionuclides, the individual would continue to accumulate dose in subsequent years if he continued to live at the same location.

C.2.2.6 Ingestion Dose

Ingestion doses were calculated using total integrated ground deposition and appropriate dose conversion factors from Tables C-4 and C-6. Methods for

evaluating the food chain pathway are taken from NRC Regulatory Guide 1.109 (NRC, 1977). Site-specific agricultural parameters for these calculations are based on those used for previous assessments for facilities at the INEL (DOE, 1987). A summary of these site-specific, nuclide independent parameter values appears in Table C-7.

Nuclide dependent parameters, such as those describing uptake by forage and crop plants and transfer to meat and milk, are taken from the scientific literature developed to update NRC Regulatory Guide 1.109 (Baes et al., 1984). The referenced parameter values also provide the basis for the data base prescribed by the EPA for dose and risk assessment of radionuclide emissions to air (EPA, 1990). One nuclide-specific parameter is used that does not appear in the original formulation of the Regulatory Guilde 1.109 model but is commonly used in upgrades such as the version prescribed by the EPA. This parameter is the translocation factor from surface deposition on plant leaves to the edible portion of the crop. Translocation factors for fresh leafy vegetables are set to 1. Translocation factors to the edible portions of produce crops were taken from Boone et al. (1981), as developed for the *Environmental Assessment: Fuel Processing Restoration at the Idaho National Engineering Laboratory* (DOE, 1987). Some of the values developed for the environmental assessment were estimated from chemically similar elements. Where not available from the environmental assessment, translocation factors were set to 1. A summary of nuclide dependent uptake and transfer parameters is presented in Table C-8.

For each radionuclide and age group, j, ingestion doses are given by

$$D_i = TIC\ V_d\ IUF_j\ DCF_j\ (1.00 \times 10^6)\ (3.60 \times 10^3)\ (1.00 \times 10^3) \qquad \text{(C-8)}$$

where

D_i = ingestion dose (mrem)

TIC = total integrated air concentration (μCi-h/m^3)

V_d = deposition velocity (m/s)

IUF_j = ingestion uptake factor (μCi per $\mu Ci/m^2$)

DCF_j = ingestion dose conversion factor (rem/μCi)

1.00×10^6 = unit conversion (Ci to μCi)

3.60×10^3 = unit conversion (h to s)

1.00×10^3 = unit conversion (mrem to rem).

Table C-7. Age-specific dietary usage parameters

Parameter	Value	Units	Symbol[a]
Annual dietary ingestion rates			
Adult			
Produce	5.2E+2	kg/yr	U_v
Milk	3.1E+2	L/yr	U_m
Meat	1.1E+2	kg/yr	U_f
Leafy vegetables	6.4E+1	kg/yr	U_l
Child (10 years)			
Produce	5.2E+2	kg/yr	U_v
Milk	3.3E+2	L/yr	U_m
Meat	4.1E+1	kg/yr	U_f
Leafy vegetables	2.6E+1	kg/yr	U_l
Infant (0-12 months)			
Produce	8.5E+1	kg/yr	U_v
Milk	3.3E+2	L/yr	U_m
Meat	1.6E+1	kg/yr	U_f
Leafy vegetables	1.0E+0	kg/yr	U_l

a. Symbols used in equations appearing in Appendix C for calculating ingestion uptake factor.

Table C-8. Nuclide independent input parameters for calculating ingestion uptake factors

Parameter	Value	Units	Symbol[a]
Fraction from sources outside area			--[b]
Vegetables	0.0	--	
Meat	0.0	--	
Milk	0.0	--	
Fraction grown in garden			--[b]
Produce	1.00	--	
Leafy vegetables	1.00	--	
Time delays			t_h
Ingestion of pasture	0.00	h	
Ingestion of feed	2.16E+3	h	
Fresh vegetables	2.40E+1	h	
Produce	1.44E+3	h	
Feed-milk-person	4.80E+1	h	
Slaughter-consumption	4.80E+2	h	
Time periods for exposure to deposition			t_b
Soil	1.32E+5	h	
Crop productivity			Y_v
Pasture	7.00E-1	kg/m^2	
Produce	2.00E+0	kg/m^2	
Fraction of year grazed (operational only)	0.40	--	f_p
Pasture fraction of feed	0.43	--	f_s
Animal feeding rate	1.56E+1	kg/d	Q_F
Soil surface density (15 cm deep)	2.25E+2	kg/m^2	P

Table C-8. (continued)

Parameter	Value	Units	Symbol[a]
Fallout interception fractions			r
Pasture	0.57	--	
Vegetables	0.20	--	
Produce	0.20	--	
Removal rate constant for weathering from plants	2.10E-3	h^{-1}	λ_w
Deposition velocities for ground surfaces			v_d
Noble gases	0.00E+0	m/s	
Halogens (elemental)	1.80E-2	m/s	
Halogens (organic)	1.80E-4	m/s	
Particulates	1.80E-3	m/s	

a. Symbols used in equations appearing in Appendix C for calculating ingestion uptake factor.

b. Parameter does not appear explicitly in equations but represents an underlying assumption in the calculations.

The ingestion uptake factor (IUF) is calculated for this evaluation using the food chain model contained in NRC Regulatory Guide 1.109 (NRC, 1977). The IUF is the estimated radionuclide ingestion (μCi) per unit integrated air concentration (μCi-h/m^3) normalized for a unit deposition rate (1 m/s). It includes estimates of radionuclide intake through consumption of milk, meat, leafy vegetables, and other produce.

The Regulatory Guide 1.109 model is a simplified treatment of the transfer of radionuclides through the food chain. It is designed to estimate the steady-state concentrations of radionuclides in foods for a constant deposition rate. One of its basic assumptions is that the transfer of atmospherically deposited radioactivity is independent of when and over what length of time the deposition occurs. One consequence of this assumption is

that the steady state concentration per unit deposition rate is equivalent to the time integrated concentration per unit of acute deposition (Whicker et al., 1990). This permits estimates of dose commitments to be scaled to total integrated concentration whether the release occurred over the entire year or over a shorter period of time (ICRP, 1978).

In this evaluation, estimates of ingestion dose for operational releases used total integrated concentrations calculated by Equation (C-1). Estimates of ingestion dose for episodic releases use total integrated concentrations calculated by Equation (C-3).

For each radionuclide and age group, j, the IUF is given by

$$IUF_j = (f_v U_{vj} C_v + f_l U_{lj} C_l + U_{fj} C_f + U_{mj} C_m) \; 1.14 \times 10^{-4} \qquad \text{(C-9)}$$

where

IUF_j	=	ingestion uptake factor (μCi per μCi/m^2)
f_v, f_l	=	fractions of dietary produce and leafy vegetables produced in a home garden
U_{vj}, Ul_{lj}, U_{fj}, U_{mj}	=	dietary usage factors for produce, leafy vegetables, meat (kg/yr), and milk (L/yr)
C_v, C_l, C_f, C_m	=	normalized concentrations in produce, leafy vegetables, meat (μCi-h/kg per μCi/m^2), and milk (μCi-h/L per μCi/m^2)
1.14×10^{-4}	=	unit conversion (h to yr).

Ingestion doses are estimated based on dietary usage factors for different age groups (i.e., adults, children, infants). Dietary intakes for all age groups were derived from those recommended in NCRP Report 76 for estimates of ingestion dose to the maximally exposed individual (NCRP, 1984).

The diets for adults and children correspond precisely to those used in NRC Regulatory Guide 1.109 for the maximally exposed individual. The NRC Regulatory Guide 1.109 recommendation for the diet of infants consists entirely of milk. For this analysis the assumed diet of infants was supplemented with meat and vegetables in amounts corresponding to the recommendations of NCRP 76 for a "best estimate" of average daily intake. Age specific dietary usage factors are summarized in Table C-7.

The normalized, time integrated concentrations of each radionuclide in produce, leafy vegetables, and forage for livestock consumption are a function of deposition on the surfaces of plant foliage and uptake from contaminated soil. The following equation was used for estimating radionuclide concentration in plants:

$$C = \left(\frac{r\ T\ (1-e^{-\lambda_E t_e})}{Y_v\ \lambda_E} + \frac{B_{iv}(1-e^{-\lambda_r t_b})}{P\ \lambda_r} \right) e^{-\lambda_r t_h} \qquad \text{(C-10)}$$

where

C	=	normalized, integrated concentration in produce, leafy vegetables, or forage (μCi-h/kg per μCi/m^2)
r	=	fraction of deposited radioactivity retained on foliage
T	=	fraction of deposited radioactivity translocated from plant surface to edible portion of crop
λ_E	=	effective removal rate constant from foliage surfaces (h^{-1}), $\lambda_E = \lambda_r + \lambda_w$
λ_r	=	radioactive decay rate constant (h^{-1})
λ_w	=	weathering rate constant (h^{-1})

t_e = time period crops are exposed to deposition (h)

Y_v = agricultural productivity (kg/m^2)

B_{iv} = concentration factor for root uptake to edible portion of crop (pCi/kg_{plant} per pCi/kg_{soil})

t_b = time period soil is exposed to deposition (h)

P = surface density of soil (kg/m^2)

t_h = time between harvest and consumption by humans or livestock (h).

The concentrations in vegetable produce, C_v, and leafy vegetables, C_1, for human consumption are calculated using appropriate values from Tables C-8 and C-9.

Both the concentration in milk, C_m, and the concentration in meat, C_f, depend on the radionuclide concentration of pasture and stored feed and on the fraction of the year that livestock is grazing on pasture. The equation used to estimate the normalized, time integrated radionuclide concentration in livestock feed is

$$C_{fd} = f_p f_s C_p + (1 - f_p f_s) C_s \qquad \text{(C-11)}$$

where

C_{fd} = normalized, integrated concentration in livestock feed (μCi-h/kg per $\mu Ci/m^2$)

f_p = fraction of the year that animals graze on pasture

f_s = fraction of feed that is forage when animals are grazing

Table C-9. Nuclide dependent input parameters for calculating ingestion uptake factors

Nuclide	Translocation Factors		Root Uptake Factors		Transfer Coefficients	
	Forage[a] T	Produce T	Forage B_{iv1}	Produce B_{iv2}	Meat F_f	Milk F_m
H-3					1.2E-02	1.0E-02
C-14					3.1E-02	1.2E-02
Na-24	1.00E+00	1.00E+00	7.5E-02	2.4E-02	5.5E-02	3.5E-02
Sc-46	1.00E+00	1.00E+00	6.0E-03	4.3E-04	1.5E-02	5.0E-06
Cr-51	1.00E+00	1.00E+00	7.5E-03	1.9E-03	5.5E-03	1.5E-03
Mn-54	1.00E+00	1.00E+00	2.5E-01	2.1E-02	4.0E-04	3.5E-04
Mn-56	1.00E+00	1.00E+00	2.5E-01	2.1E-02	4.0E-04	3.5E-04
Co-58	1.00E+00	1.00E+00	2.0E-02	3.0E-03	2.0E-02	2.0E-03
Co-60	1.00E+00	1.00E+00	2.0E-02	3.0E-03	2.0E-02	2.0E-03
As-76	1.00E+00	1.00E+00	4.0E-02	2.6E-03	2.0E-03	6.0E-05
Br-82	1.00E+00	1.00E+00	1.5E+00	6.4E-01	2.5E-02	2.0E-02
Br-84	1.00E+00	1.00E+00	1.5E+00	6.4E-01	2.5E-02	2.0E-02
Rb-88	1.00E+00	1.00E+00	1.5E-01	3.0E-02	1.5E-02	1.0E-02
Rb-89	1.00E+00	1.00E+00	1.5E-01	3.0E-02	1.5E-02	1.0E-02
Rb-90	1.00E+00	1.00E+00	1.5E-01	3.0E-02	1.5E-02	1.0E-02
Sr-89	1.00E+00	1.00E-01	2.5E+00	1.1E-01	3.0E-04	1.5E-03
Sr-90	1.00E+00	1.00E-01	2.5E+00	1.1E-01	3.0E-04	1.5E-03
Sr-91	1.00E+00	1.00E-01	2.5E+00	1.1E-01	3.0E-04	1.5E-03
Sr-92	1.00E+00	1.00E-01	2.5E+00	1.1E-01	3.0E-04	1.5E-03
Y-91m	1.00E+00	1.00E-01	1.5E-02	2.6E-03	3.0E-04	2.0E-05
Y-92	1.00E+00	1.00E-01	1.5E-02	2.6E-03	3.0E-04	2.0E-05
Y-93	1.00E+00	1.00E-01	1.5E-02	2.6E-03	3.0E-04	2.0E-05
Y-94	1.00E+00	1.00E-01	1.5E-02	2.6E-03	3.0E-04	2.0E-05
Y-95	1.00E+00	1.00E-01	1.5E-02	2.6E-03	3.0E-04	2.0E-05
Nb-95	1.00E+00	1.00E+00	2.0E-02	2.1E-03	2.5E-01	2.0E-02
Nb-95m	1.00E+00	1.00E+00	2.0E-02	2.1E-03	2.5E-01	2.0E-02
Nb-96	1.00E+00	1.00E+00	2.0E-02	2.1E-03	2.5E-01	2.0E-02
Nb-97	1.00E+00	1.00E+00	2.0E-02	2.1E-03	2.5E-01	2.0E-02
Zr-95	1.00E+00	2.00E-02	2.0E-03	2.1E-04	5.5E-03	3.0E-05
Zr-97	1.00E+00	2.00E-02	2.0E-03	2.1E-04	5.5E-03	3.0E-05
Mo-99	1.00E+00	1.00E+00	2.5E-01	2.6E-02	6.0E-03	1.5E-03
Mo-101	1.00E+00	1.00E+00	2.5E-01	2.6E-02	6.0E-03	1.5E-03
Tc-99m	1.00E+00	1.00E+00	9.5E+00	6.4E-01	8.5E-03	1.0E-02
Tc-101	1.00E+00	1.00E+00	9.5E+00	6.4E-01	8.5E-03	1.0E-02
Tc-104	1.00E+00	1.00E+00	9.5E+00	6.4E-01	8.5E-03	1.0E-02
Ru-103	1.00E+00	5.00E-02	7.5E-02	8.6E-03	2.0E-03	6.0E-07
Ru-105	1.00E+00	5.00E-02	7.5E-02	8.6E-03	2.0E-03	6.0E-07
Ru-106	1.00E+00	5.00E-02	7.5E-02	8.6E-03	2.0E-03	6.0E-07
Rh-105	1.00E+00	5.00E-02	1.5E-01	1.7E-02	2.0E-03	1.0E-02

Table C-9. (continued)

Nuclide	Translocation Factors: Forage[a] T	Translocation Factors: Produce T	Root Uptake Factors: Forage B_{iv1}	Root Uptake Factors: Produce B_{iv2}	Transfer Coefficients: Meat F_f	Transfer Coefficients: Milk F_m
Ag-111	1.00E+00	1.00E+00	4.0E-01	4.3E-02	3.0E-03	2.0E-02
In-115m	1.00E+00	1.00E+00	4.0E-03	1.7E-04	8.0E-03	1.0E-04
Sn-125	1.00E+00	1.00E+00	3.0E-02	2.6E-03	8.0E-02	1.0E-03
Sn-128	1.00E+00	1.00E+00	3.0E-02	2.6E-03	8.0E-02	1.0E-03
Sb-125	1.00E+00	2.00E-02	2.0E-01	1.3E-02	1.0E-03	1.0E-04
Sb-127	1.00E+00	2.00E-02	2.0E-01	1.3E-02	1.0E-03	1.0E-04
Sb-129	1.00E+00	2.00E-02	2.0E-01	1.3E-02	1.0E-03	1.0E-04
Te-129m	1.00E+00	2.00E-02	2.5E-02	1.7E-03	1.5E-02	2.0E-04
Te-131	1.00E+00	2.00E-02	2.5E-02	1.7E-03	1.5E-02	2.0E-04
Te-131m	1.00E+00	2.00E-02	2.5E-02	1.7E-03	1.5E-02	2.0E-04
Te-132	1.00E+00	2.00E-02	2.5E-02	1.7E-03	1.5E-02	2.0E-04
Te-133	1.00E+00	2.00E-02	2.5E-02	1.7E-03	1.5E-02	2.0E-04
Te-134	1.00E+00	2.00E-02	2.5E-02	1.7E-03	1.5E-02	2.0E-04
I-129	1.00E+00	1.00E-01	1.5E-01	2.1E-02	7.0E-03	1.0E-02
I-131	1.00E+00	1.00E-01	1.5E-01	2.1E-02	7.0E-03	1.0E-02
I-132	1.00E+00	1.00E-01	1.5E-01	2.1E-02	7.0E-03	1.0E-02
I-133	1.00E+00	1.00E-01	1.5E-01	2.1E-02	7.0E-03	1.0E-02
I-134	1.00E+00	1.00E-01	1.5E-01	2.1E-02	7.0E-03	1.0E-02
I-135	1.00E+00	1.00E-01	1.5E-01	2.1E-02	7.0E-03	1.0E-02
Cs-134	1.00E+00	3.50E-01	8.0E-02	1.3E-02	2.0E-02	7.0E-03
Cs-137	1.00E+00	3.50E-01	8.0E-02	1.3E-02	2.0E-02	7.0E-03
Cs-138	1.00E+00	3.50E-01	8.0E-02	1.3E-02	2.0E-02	7.0E-03
Cs-139	1.00E+00	3.50E-01	8.0E-02	1.3E-02	2.0E-02	7.0E-03
Ba-139	1.00E+00	1.00E-01	1.5E-01	6.4E-03	1.5E-04	3.5E-04
Ba-140	1.00E+00	1.00E-01	1.5E-01	6.4E-03	1.5E-04	3.5E-04
Ba-141	1.00E+00	1.00E-01	1.5E-01	6.4E-03	1.5E-04	3.5E-04
Ba-142	1.00E+00	1.00E-01	1.5E-01	6.4E-03	1.5E-04	3.5E-04
La-140	1.00E+00	1.00E+00	1.0E-02	1.7E-03	3.0E-04	2.0E-05
La-141	1.00E+00	1.00E+00	1.0E-02	1.7E-03	3.0E-04	2.0E-05
La-142	1.00E+00	1.00E+00	1.0E-02	1.7E-03	3.0E-04	2.0E-05
La-143	1.00E+00	1.00E+00	1.0E-02	1.7E-03	3.0E-04	2.0E-05
Ce-141	1.00E+00	2.00E-02	1.0E-02	1.7E-03	7.5E-04	2.0E-05
Ce-143	1.00E+00	2.00E-02	1.0E-02	1.7E-03	7.5E-04	2.0E-05
Ce-144	1.00E+00	2.00E-02	1.0E-02	1.7E-03	7.5E-04	2.0E-05
Pr-143	1.00E+00	2.00E-02	1.0E-02	1.7E-03	3.0E-04	2.0E-05
Pr-144	1.00E+00	2.00E-02	1.0E-02	1.7E-03	3.0E-04	2.0E-05
Pr-147	1.00E+00	2.00E-02	1.0E-02	1.7E-03	3.0E-04	2.0E-05
Pm-147	1.00E+00	2.00E-02	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Pm-149	1.00E+00	2.00E-02	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Pm-151	1.00E+00	2.00E-02	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Nd-147	1.00E+00	1.00E+00	1.0E-02	1.7E-03	3.0E-04	2.0E-05

Table C-9. (continued)

Nuclide	Translocation Factors: Forage[a] T	Translocation Factors: Produce T	Root Uptake Factors: Forage B_{iv1}	Root Uptake Factors: Produce B_{iv2}	Transfer Coefficients: Meat F_f	Transfer Coefficients: Milk F_m
Nd-149	1.00E+00	1.00E+00	1.0E-02	1.7E-03	3.0E-04	2.0E-05
Sm-153	1.00E+00	1.00E+00	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Eu-154	1.00E+00	2.00E-02	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Eu-155	1.00E+00	2.00E-02	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Eu-156	1.00E+00	2.00E-02	1.0E-02	1.7E-03	5.0E-03	2.0E-05
Ta-182	1.00E+00	1.00E+00	1.0E-02	1.1E-03	6.5E-04	3.0E-06
Ir-192	1.00E+00	1.00E+00	5.5E-02	6.4E-03	1.5E-03	2.0E-06
Hg-203	1.00E+00	1.00E+00	9.0E-01	8.6E-02	2.5E-01	4.5E-04
Th-232	1.00E+00	1.00E+00	8.5E-04	3.6E-05	6.0E-06	5.0E-06
U-234	1.00E+00	1.00E+00	8.5E-03	1.7E-03	2.0E-04	6.0E-04
U-235	1.00E+00	1.00E+00	8.5E-03	1.7E-03	2.0E-04	6.0E-04
U-238	1.00E+00	1.00E+00	8.5E-03	1.7E-03	2.0E-04	6.0E-04
Np-237	1.00E+00	1.00E+00	1.0E-01	4.3E-03	5.5E-05	5.0E-06
Pu-238	1.00E+00	1.00E-02	4.5E-04	1.9E-05	5.0E-07	1.0E-07
Pu-239	1.00E+00	1.00E-02	4.5E-04	1.9E-05	5.0E-07	1.0E-07
Pu-240	1.00E+00	1.00E-02	4.5E-04	1.9E-05	5.0E-07	1.0E-07

a. Translocation factors to edible portions of leafy vegetables and forage

C_p = normalized, integrated concentration on forage (μCi-h/kg per μCi/m^2)

C_s = normalized, integrated concentration in stored feed (μCi-h/kg per μCi/m^2).

Both C_p and C_s are calculated using Equation (C-10) with values from Tables C-8 and C-9 appropriate to forage and stored feed. One parameter in Equation (C-10) requires a separate calculation of IUFs for episodic and operational releases. This is f_s, the fraction of the year that animals graze on pasture. For operational releases, it is appropriate to use a value that reflects the average annual grazing time of all cattle in the region (0.40). For short term releases, where it is not known whether or not cattle are actually grazing during the release, it is more appropriate to assume f_s is

equal to 1.0. This results in a greater fraction of deposited material available for incorporation into the meat and milk of grazing cattle for episodic releases. Consequently, the IUFs for episodic releases are larger in proportion to the dietary contributions of meat and milk.

The normalized radionuclide concentration in milk for human consumption is given by

$$C_m = F_m C_{fd} Q_f e^{-\lambda_r t_f} \qquad \text{(C-12)}$$

where

C_m = normalized, integrated concentration in milk (μCi-h/L per μCi/m^2)

F_m = transfer coefficient from feed to milk (d/L)

C_{fd} = normalized, integrated concentration in livestock feed (μCi-h/kg per μCi/m^2)

Q_F = consumption rate of feed by animal (kg/d)

λ_r = radioactive decay rate constant (h^{-1})

t_f = time between deposition and consumption of milk (h).

Similarly, the radionuclide normalized, integrated concentration in meat for human consumption is given by

$$C_f = F_d C_{fd} Q_f e^{-\lambda_r t_b} \qquad \text{(C-13)}$$

where

C_f = normalized, integrated concentration in meat (μCi-h/kg per μCi/m^2)

F_d = transfer coefficient from feed to meat (d/kg)

C_{fd} = normalized, integrated concentration in livestock feed (μCi-h/kg per μCi/m^2)

Q_f = consumption rate of feed by animal (kg/d)

λ_r = radioactive decay rate constant (h^{-1})

t_b = time between slaughter and consumption of meat (h).

The Regulatory Guide 1.109 model does not take into account seasonal variations that might affect radionuclide transfer to the food chain for short-term releases. Developing a more sophisticated model that accounts for all the seasonal characteristics of Idaho is outside the scope of this study. However, in Idaho there are seasons of the year when no mechanism exists for deposition of radionuclides onto food crops (e.g., no food crops or forage grow during Idaho's winters). Radionuclides with short half-lives released for only a few days during the winter will not enter the human food chain by direct deposition on crops or livestock forage. However, ingestion of short-lived radionuclides may occur following winter time releases. Models have demonstrated that indirect pathways, such as consumption of contaminated dirt and inhalation of contaminated air by dairy cattle, have some potential for introducing a reduced amount of radioactivity into the food chain during winter months. For example, the transfer of I-131 to milk from fallout following atmospheric weapons tests during winter months could have been as much as 10% of the summertime value (Whicker et al., 1990).

It is unlikely that any significant deposition onto forage or crops could occur between early November and late April in Southeastern Idaho. However, for this analysis it was conservatively assumed that the ingestion pathway would be reduced for those months the ground surface is typically frozen. Data reported by National Oceanic and Atmospheric Administration

(NOAA) indicate that the average daily high temperatures at 1 in. soil depth do not exceed freezing on the INEL Site during December, January, and February (Clawson et al., 1989). Therefore, the IUF for all radionuclides with half-lives of less than 10 days was set to 10% of its summertime value for all episodic releases occurring during these 3 months. The seven releases in this category are SNAPTRAN-2 (1/11/66), CERT No. 3 (12/11/64), 1961 ICPP Criticality (1/25/61), SL-1 Accident (1/3/61), IET #18 (1/6 through 2/5/60), IET #6 (12/18/56), and IET #3 (2/11/56). Some of the extended episodic releases occurred over several weeks and overlapped the winter period. In these cases, wintertime IUFs were used only for the portion of the releases occurring during the months of December, January, and February. The four releases in this category are IET #26 (12/23/60 through 3/30/61), IET #25 (11/22 through 12/15/60), IET #17 (11/2 through 12/12/59), and IET #10 (12/20/57 through 3/6/58). No seasonal adjustments were made to the IUFs used for operational releases even though some portion of each annual release occurred during the winter months.

Values for the IUFs calculated using the equations and parameter values presented in this section are listed in Tables C-10 and C-11. Values for operational releases appear in Table C-10 and values for episodic releases appear in C-11.

Table C-10. Ingestion uptake factors for operational releases (μCi per μCi/m^2)

Radionuclide	Ingestion Uptake Factor		
	Adult	Infant	Child
H-3	7.98E+04	1.72E+04	7.39E+04
C-14	4.78E+05	1.07E+05	4.31E+05
N-13	0.00E+00	0.00E+00	0.00E+00
Na-24	1.11E-02	6.57E-03	8.53E-03
Ar-41	0.00E+00	0.00E+00	0.00E+00
Sc-46	2.11E+00	2.96E-01	1.70E+00
Cr-51	7.28E-01	1.28E-01	5.73E-01
Mn-54	2.72E+00	4.41E-01	2.52E+00
Mn-56	4.27E-06	6.69E-08	1.73E-06
Co-58	2.16E+00	4.41E-01	1.72E+00
Co-60	4.48E+00	9.59E-01	3.58E+00
As-76	1.37E-02	2.63E-04	5.60E-03
Br-82	4.96E-02	3.01E-02	3.86E-02
Kr-85	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00
Kr-90	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.31E-32	3.62E-34	9.40E-33
Rb-90	0.00E+00	0.00E+00	0.00E+00
Sr-89	4.86E-01	1.23E-01	3.29E-01
Sr-90	5.53E+00	1.93E+00	5.12E+00
Sr-91	1.75E-03	7.69E-05	7.41E-04
Y-91	6.94E-03	2.45E-02	2.24E-01
Y-91m	1.67E-12	2.60E-14	6.77E-13
Zr-95	4.15E-01	2.69E-02	1.80E-01
Nb-95	4.18E+00	1.33E+00	2.51E+00
Tc-99m	4.32E-04	3.58E-05	1.94E-04
Ru-103	3.10E-01	1.34E-02	1.47E-01
Ru-106	5.65E-01	4.28E-02	3.11E-01
Sb-125	5.83E-01	5.93E-02	3.43E-01
Te-132	5.53E-02	1.90E-03	2.31E-02
I-129	4.11E+00	2.54E+00	3.66E+00
I-131	2.37E-01	1.23E-01	1.71E-01
I-132	1.75E-06	2.87E-08	7.10E-07
I-133	1.37E-02	4.84E-03	8.47E-03
I-134	5.32E-12	8.31E-14	2.16E-12
I-135	6.00E-04	5.89E-05	2.75E-04
Xe-131m	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00
Xe-137	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00
Xe-139	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.52E+00	1.51E+00	2.69E+00
Cs-136	3.39E-01	1.33E-01	2.23E-01
Cs-137	4.34E+00	1.81E+00	3.36E+00
Cs-138	1.96E-17	3.07E-19	7.97E-18
Cs-139	1.30E-50	2.02E-52	5.26E-51
Ba-139	8.86E-09	1.38E-10	3.60E-09
Ba-140	1.70E-01	9.28E-03	7.57E-02
Ce-141	2.55E-01	7.28E-03	1.10E-01
Ce-144	4.03E-01	2.08E-02	1.93E-01
Pr-143	1.66E-01	3.23E-03	6.85E-02
Pm-147	6.72E-01	5.92E-02	3.04E-01
Eu-154	7.17E-01	6.55E-02	3.35E-01

Table C-10. (continued)

Radionuclide	Ingestion Uptake Factor		
	Adult	Infant	Child
Eu-155	6.98E-01	6.28E-02	3.21E-01
Ta-182	2.02E+00	2.85E-01	1.83E+00
Ir-192	1.64E+00	2.26E-01	1.45E+00
Hg-203	4.91E+00	7.19E-01	2.42E+00
Th-232	3.02E+00	4.46E-01	2.83E+00
Np-237	3.19E+00	4.71E-01	2.98E+00
U-238	3.20E+00	5.65E-01	3.00E+00
Pu-238	3.58E-01	9.68E-03	1.62E-01
Pu-239	3.59E-01	9.70E-03	1.62E-01
Pu-240	3.59E-01	9.70E-03	1.62E-01
Pu-239/240	3.59E-01	9.70E-03	1.62E-01

Table C-11. Ingestion uptake factors for episodic releases (μCi per μCi/m^2)

Radionuclide	Ingestion Uptake Factor		
	Adult	Infant	Child
Br-84	1.35E-17	2.11E-19	5.48E-18
Kr-85m	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.31E-32	3.62E-34	9.40E-33
Sr-89	5.30E-01	1.68E-01	3.73E-01
Sr-90	5.53E+00	1.93E+00	5.12E+00
Sr-91	1.82E-03	1.52E-04	8.16E-04
Sr-92	6.16E-06	1.02E-07	2.51E-06
Y-91	9.51E-03	2.53E-02	2.26E-01
Y-92	3.36E-05	5.27E-07	1.37E-05
Y-93	1.99E-03	3.33E-05	8.11E-04
Zr-95	4.53E-01	3.30E-02	1.95E-01
Zr-97	6.35E-03	1.19E-04	2.59E-03
Nb-96	4.06E-02	3.13E-02	3.57E-02
Mo-99	5.93E-02	1.56E-02	3.32E-02
Ru-103	3.24E-01	1.55E-02	1.52E-01
Ru-105	1.09E-04	1.71E-06	4.43E-05
Ru-106	5.69E-01	4.34E-02	3.12E-01
Sb-129	1.05E-04	1.70E-06	4.25E-05
Te-131	2.03E-21	3.17E-23	8.23E-22
Te-131m	1.72E-02	7.99E-04	7.31E-03
Te-132	5.75E-02	3.49E-03	2.48E-02
Te-133m	1.46E-11	2.28E-13	5.93E-12
Te-134	3.14E-14	4.91E-16	1.28E-14
I-131	4.15E-01	3.04E-01	3.54E-01
I-132	1.75E-06	3.08E-08	7.12E-07
I-133	2.03E-02	1.19E-02	1.55E-02
I-134	5.32E-12	8.31E-14	2.16E-12
I-135	6.70E-04	1.34E-04	3.50E-04
Xe-129m	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00
Cs-137	4.34E+00	1.82E+00	3.37E+00
Cs-138	1.96E-17	3.07E-19	7.97E-18
Ba-139	8.86E-09	1.38E-10	3.60E-09
Ba-140	1.78E-01	1.78E-02	8.44E-02
Ba-141	6.09E-28	9.52E-30	2.48E-28
Ba-142	5.89E-45	9.21E-47	2.39E-45
La-141	6.03E-05	9.47E-07	2.45E-05
La-142	4.78E-08	7.47E-10	1.94E-08
Ce-141	2.60E-01	8.61E-03	1.13E-01
Ce-143	1.92E-02	3.64E-04	7.83E-03
Ce-144	4.06E-01	2.13E-02	1.94E-01
Pr-143	1.68E-01	3.87E-03	6.93E-02
Pr-144	2.52E-29	3.94E-31	1.02E-29
U-234	3.20E+00	5.65E-01	3.00E+00
U-235	3.20E+00	5.65E-01	3.00E+00
U-238	3.20E+00	5.65E-01	3.00E+00

C.3 EXAMPLE DOSE CALCULATIONS

The methods for radiation dose calculation described in this report are best illustrated by working through an example calculation. The example presented in this section demonstrates how to estimate the total annual dose to the thyroid of a maximally exposed infant because of releases of I-131. The EDE attributable to I-131 is also estimated. The example calculation indicates where each equation and parameter value can be found in this report. This allows the reader to perform similar calculations for other operational or episodic releases, radionuclides, locations, organs, or age groups if desired.

The example calculation consists of three parts: (1) estimating dose from annual operational releases, (2) estimating dose from an episodic release occurring during that year, and (3) combining the two estimates to determine the total for the location at which the highest dose was received.

The equations used for this example calculation appear in Section C.2.2 along with definitions of the parameters and their units. The values of each parameter are provided in this section.

For the purposes of illustration, operational releases for the year 1960 are used. This year provides a combination of operational and episodic releases that are simple enough to work through as an example yet complex enough to illustrate all the key features of the dose calculation method.

C.3.1 1960 Operational Releases

For each radionuclide and age group, both the operational and episodic dose calculations use the same dose conversion factors. The dosimetric and physiological parameters needed for calculating I-131 doses to an infant are presented in Table C-12. The last two columns of this table indicate the report section and table from which the value was taken.

Table C-12. Dosimetric data for I-131 (infant)

Parameter	Value	Units	Source of Data: Section Number	Source of Data: Table Number
Dose conversion factors for dose to thyroid				
Air	2.07E+3	mrem/y per $\mu Ci/m^3$	C.2.2.1	C-1
Ground	4.38E+1	mrem/y per $\mu Ci/m^2$	C.2.2.1	C-2
Inhalation	8.51E+0	rem/μCi uptake	C.2.2.1	C-5
Ingestion	1.37E+1	rem/μCi uptake	C.2.2.1	C-6
Dose conversion factors for EDE				
Air	1.91E+3	mrem/y per $\mu Ci/m^3$	C.2.2.1	C-1
Ground	4.04E+1	mrem/y per $\mu Ci/m^2$	C.2.2.1	C-2
Inhalation	2.55E-1	rem/μCi uptake	C.2.2.1	C-5
Ingestion	4.07E-1	rem/μCi uptake	C.2.2.1	C-6
Breathing rate, BR	1.58E-1	m^3/h	C.2.2.4	NA
Radioactive decay constant for I-131, λ_r	3.59E-3	h^{-1}		

The parameters needed for calculating doses from operational releases of I-131 during 1960 are presented in Table C-13. The last two columns of this table indicate the report section or table from which the value was taken.

For operational releases, the first step is calculating the total integrated concentration in air at the chosen location. The total integrated air concentration for each radionuclide is given by

$$TIC = \frac{Q_o}{T} MDF_o \, e^{-\lambda_r \frac{d}{v}} \qquad \text{(C-1)}$$

Table C-13. 1960 operational release data for I-131

Parameter	Value	Units	Source of Data: Section Number	Source of Data: Table Number
Amount released[a], Q_o	32.6	Ci	A.3.1; C.4	A-39; C-28
Maximum MDF[b], MDF_o	5.00E-08	h^2/m^3	B.4	B-1
Distance[b], d	19.1	km	B.4	B-1
Average wind speed, v	16.4	km/h	B.4	B-1
Deposition velocity, V_d	0.018	m/s	C.2.2.6	C-8
Uptake factor, IUF_i	1.23E-01	μCi per $\mu Ci/m^2$	C.2.2.6	C-10
Shielding factor, SF	0.7	--	C.2.2.5	NA

a. Releases are assumed to originate from a point midway between TRA and ICPP.

b. Data are for Atomic City.

where the parameters are those defined in Section C.2.2.2. The radioactive decay constant, λ_r, is the fraction of the radionuclide that decays in any period of time. It is equal to the natural logarithm of 2 divided by the radiological half-life.

During 1960, the location with the maximum average MDF was Atomic City. Consequently, the maximally exposed individual for operational doses will be located at Atomic City. Substituting appropriate values from Tables C-12 and C-13 for each parameter in Equation (C-1) gives

$$TIC = \left(\frac{32.6\ Ci}{8760\ h}\right) (5.0 \times 10^{-8}\ \frac{h^2}{m^3})\ e^{-(3.59 \times 10^{-3}\ h^{-1}) \left(\frac{19.1\ km}{16.4\ km/h}\right)}$$

$$TIC = 1.85 \times 10^{-10}\ \frac{Ci\text{-}h}{m^3}$$

This total integrated air concentration represents the releases over the entire year. The average air concentration during this same time period is given by

$$AAC = \frac{TIC}{T} \tag{C-2}$$

Substituting the calculated value of TIC into Equation (C-2) gives

$$AAC = \frac{1.85 \times 1^{-10}\ Ci\text{-}h/m^3}{8760\ h}$$

$$AAC = 2.12 \times 10^{-14}\ \frac{Ci}{m^3}$$

C.3.1.1 Air Immersion Dose

External radiation dose from immersion in contaminated air can be calculated from TIC using the following equation:

$$D_a = TIC\ DCF_a\ (1.00 \times 10^6)\ (1.14 \times 10^{-4}) \tag{C-5}$$

Substituting the appropriate dose conversion factor for the infant thyroid dose from Table C-12 gives

$$D_{a\ Thyroid} = \left(1.85 \times 10^{-10}\ \frac{Ci\text{-}h}{m^3}\right)\left(2.07 \times 10^3\ \frac{mrem/y}{\mu Ci/m^3}\right) \times \left(1.0 \times 10^6\ \frac{\mu Ci}{Ci}\right)\left(1.14 \times 10^{-4}\ \frac{y}{h}\right)$$

$$D_{a\ Thyroid} = 4.37 \times 10^{-5}\ mrem$$

Substituting the appropriate dose conversion factor for the infant EDE from Table C-12 gives

$$D_{a\ EDE} = \left(1.85 \times 10^{-10}\ \frac{Ci\text{-}h}{m^3}\right)\left(1.93 \times 10^{3}\ \frac{mrem/y}{\mu Ci/m^3}\right) \times \left(1.0 \times 10^{6}\ \frac{\mu Ci}{Ci}\right)\left(1.14 \times 10^{-4}\ \frac{y}{h}\right)$$

$$D_{a\ EDE} = 4.03 \times 10^{-5}\ mrem$$

C.3.1.2 Inhalation Dose

Internal radiation dose from inhalation of contaminated air can be calculated from the TIC if the breathing rate is known by using the equation

$$D_i = TIC\ BR_j\ DCF_{ij}\ (1.00 \times 10^{6})\ (1.00 \times 10^{3}) \qquad \text{(C-6)}$$

Substituting the breathing rate and the appropriate dose conversion factor for the infant thyroid dose from Table C-12 gives

$$D_{i\ Thyroid} = \left(1.85 \times 10^{-10}\ \frac{Ci\text{-}h}{m^3}\right)\left(1.58 \times 10^{-1}\ \frac{m^3}{h}\right)\left(8.51\ \frac{rem}{\mu Ci}\right) \times \left(1.00 \times 10^{6}\ \frac{\mu Ci}{Ci}\right)\left(1.00 \times 10^{3}\ \frac{mrem}{rem}\right)$$

$$D_{i\ Thyroid} = 2.49 \times 10^{-1}\ mrem$$

Substituting the breathing rate and the appropriate dose conversion factor for the infant EDE from Table C-12 gives

$$D_{i\ EDE} = \left(1.85 \times 10^{-10}\ \frac{Ci\text{-}h}{m^3}\right)\left(1.58 \times 10^{-1}\ \frac{m^3}{h}\right)\left(2.55 \times 10^{-1}\ \frac{rem}{\mu Ci}\right) \times \left(1.00 \times 10^{6}\ \frac{\mu Ci}{Ci}\right)\left(1.00 \times 10^{3}\ \frac{mrem}{rem}\right)$$

$$D_{i\ EDE} = 7.45 \times 10^{-3}\ mrem$$

C.3.1.3 Ground Surface Dose

Calculating external radiation dose from contaminated ground surfaces is much more complex. It requires estimating radioactive decay over the period of deposition and during the 50-year dose commitment period following deposition. This introduces two exponential terms into the equation. Calculating radioactive decay during deposition also requires estimating average deposition rate during the period of exposure. This introduces a factor containing the average air concentration (AAC) and the deposition velocity (V_d). Dose from contaminated ground surfaces is given by

$$D_S = AAC\ V_d\ SF\ DCF_s\ \frac{1}{\lambda_r}\left(t - \frac{e^{-\lambda_r T}}{\lambda_r}(1-e^{-\lambda_r t})\right) \times \qquad \text{(C-7)}$$

$$(3.60 \times 10^3)\ (1.00 \times 10^6)\ (1.14 \times 10^{-4})$$

To simplify this equation, the exponential terms will be evaluated first. For operational releases the period of exposure, t, is 1 year (8760 h) and the dose commitment period, T, is 50 years (4.38×10^5 h). Substituting in the bracketed expression from the equation above gives

$$t - \frac{e^{-\lambda_r T}}{\lambda_r}(1-e^{-\lambda_r t}) =$$

$$(8760h) - \frac{e^{-(3.59 \times 10^{-3}\ h^{-1})(4.38 \times 10^5\ h)}}{3.59 \times 10^{-3} h^{-1}}(1-e^{-(3.59 \times 10^{-3} h^{-1})(8760\ h)}) \approx$$

$$(8760\ h) - \frac{0}{3.59 \times 10^{-3} h^{-1}}(1-0) = 8760\ h$$

The simplification is a consequence of the 8.04 day half-life of I-131. This half-life is very short compared to both the 1 year period of exposure and the 50-year dose commitment period. Therefore, I-131 does not build up during the year of exposure or persist for an extended period following deposition.

Substituting the simplified exponential term, appropriate parameter values from Table C-13, and the dose conversion factor for the infant thyroid dose from Table C-12 gives

$$D_{S\ Thyroid} = \left(2.12 \times 10^{-14} \frac{Ci}{m^3}\right) \left(0.018 \frac{m}{s}\right) (0.7) \left(4.38 \times 10^{1} \frac{mrem/y}{\mu Ci/m^2}\right) \times$$

$$\left(\frac{8760h}{3.59 \times 10^{-3} h^{-1}}\right) \left(3.60 \times 10^{3} \frac{s}{h}\right) \left(1.00 \times 10^{6} \frac{\mu Ci}{Ci}\right) \left(1.14 \times 10^{-4} \frac{y}{h}\right)$$

$$D_{s\ Thyroid} = 1.17 \times 10^{-2}\ mrem$$

Substituting the simplified exponential term, appropriate parameter values from Table C-13, and the dose conversion factor for the infant EDE from Table C-12 gives

$$D_{S\ EDE} = \left(2.12 \times 10^{-14} \frac{Ci}{m^3}\right) \left(0.018 \frac{m}{s}\right) (0.7) \left(4.04 \times 10^{1} \frac{mrem/y}{\mu Ci/m^2}\right) \times$$

$$\left(\frac{8760h}{3.59 \times 10^{-3} h^{-1}}\right) \left(3.60 \times 10^{3} \frac{s}{h}\right) \left(1.00 \times 10^{6} \frac{\mu Ci}{Ci}\right) \left(1.14 \times 10^{-4} \frac{y}{h}\right)$$

$$D_{s\ EDE} = 1.08 \times 10^{-2}\ mrem$$

C.3.1.4 Ingestion Dose

Calculation of ingestion doses involves a series of equations that estimate, for each radionuclide, the fraction of deposited material that is ultimately consumed in the diet. This fraction is called the ingestion uptake factor (IUF) and is calculated using Equations (C-9) through (C-13) as described in Section C.2.2.6. The IUF differs for each radionuclide according to chemical differences affecting its behavior in the environment. The IUF differs among age groups because of differences in diet. It also differs

between operational and episodic releases because of differences in treatment of grazing cattle for the two situations. Ingestion doses are given by

$$D_i = TIC\ V_d\ IUF\ DCF_i\ (1.00 \times 10^6)\ (3.60 \times 10^3)\ (1.00 \times 10^3) \qquad \text{(C-8)}$$

Substituting appropriate parameter values and the dose conversion factor for the infant thyroid dose from Table C-12 into Equation (C-8) gives

$$D_{i\ Thyroid} = \left(1.85 \times 10^{-10} \frac{Ci\text{-}h}{m^3}\right)\left(1.8 \times 10^{-2} \frac{m}{s}\right)\left(1.23 \times 10^{-1} \frac{\mu Ci}{\mu Ci/m^2}\right) \times$$

$$\left(1.37 \times 10^{+1} \frac{rem}{\mu Ci}\right)\left(1.00 \times 10^6 \frac{\mu Ci}{Ci}\right)\left(3.60 \times 10^3 \frac{s}{h}\right)\left(1.00 \times 10^3 \frac{mrem}{rem}\right)$$

$$D_{i\ Thyroid} = 2.02 \times 10^1\ mrem$$

Substituting appropriate parameter values and the dose conversion factor for the infant EDE from Table C-12 into Equation (C-8) gives

$$D_{i\ EDE} = \left(1.85 \times 10^{-10} \frac{Ci\text{-}h}{m^3}\right)\left(1.8 \times 10^{-2} \frac{m}{s}\right)\left(1.23 \times 10^{-1} \frac{\mu Ci}{\mu Ci/m^2}\right) \times$$

$$\left(4.07 \times 10^{-1} \frac{rem}{\mu Ci}\right)\left(1.00 \times 10^6 \frac{\mu Ci}{Ci}\right)\left(3.60 \times 10^3 \frac{s}{h}\right)\left(1.00 \times 10^3 \frac{mrem}{rem}\right)$$

$$D_{i\ EDE} = 6.00 \times 10^{-1}\ mrem$$

C.3.1.5 Total Operational Dose

The doses from all four pathways can now be added. Table C-14 presents the estimated thyroid dose and EDE for an infant at Atomic City exposed to operational releases during the year 1960. The totals for all four pathways are 20.5 mrem to the thyroid and 0.618 mrem EDE. The largest percentage of the dose results from ingestion. This is because iodine is transferred

Table C-14. Summary of I-131 doses from operational releases during 1960 for an infant residing at Atomic City

Pathway	EDE (mrem)	Thyroid (mrem)
Air immersion	4.03E-05	4.37E-05
Inhalation	7.45E-03	2.49E-01
Ground surface	1.08E-02	1.17E-02
Ingestion	6.00E-01	2.02E+01
Total	6.18E-01	2.04E+01

efficiently through the grass-cow-milk pathway and because the infant diet consists largely of milk. Other radionuclides would show different patterns of dose distribution among the four pathways depending on their half-life, physical form, chemistry, and the characteristics of their radiations.

When all radionuclides are evaluated for operational releases during 1960, the total thyroid dose is estimated to be 21.2 mrem. The I-131 dose calculated in this example accounts for more than 95% of this total. The total EDE, including all radionuclides, is estimated to be 1.41 mrem. The I-131 dose accounts for 44% of that value. External dose from the noble gases, (isotopes of Ar, Kr, and Xe) accounts for about 37% of the EDE, with the remainder distributed among the rest of the radionuclides released.

C.3.2 1960 Episodic Releases

Eleven episodic releases occurred during 1960. For this example, IET #23 (FEET #2) is presented. This release makes a significant contribution to the infant thyroid dose for 1960 and is simple enough to illustrate the calculation procedure.

The parameters needed for calculating doses from episodic releases of I-131 during 1960 are presented in Table C-15. The last two columns of this table indicate the report section or table from which the value was taken.

Table C-15. 1960 episodic release data for I-131

Parameter[a]	Value	Units	Source of Data: Section Number	Source of Data: Table Number
Amount released	4.55	Ci	A.3.1; C.4	A-41; C-34
Maximum MDF[b]	3.37E-12	h/m^3	B.4	B-2
Modeled time	310	h	B.4	B-2
Deposition velocity	0.018	m/s	C.2.2.6	C-8
IUF (infant)	3.04E-01	μCi per $\mu Ci/m^2$	C.2.2.6	C-10
Shielding factor, SF	0.7	--	C.2.2.5	NA

a. Data are for IET #23 (FEET #2) only.

b. Maximum MDF occurs at Monteview.

For operational releases, the source terms were based largely on measurements made at the point of release. They were then corrected for radioactive decay during transit to the Site boundary as part of the radiation dose calculation using Equation (C-1). This step is not necessary for the episodic release source terms. They were based on RSAC-4 computer code calculations that automatically provide a decay-corrected source term as output. The total integrated air concentration calculation for episodic releases also differs from that used for operational releases in that it does not contain a division by the time period of the release. This is because the mesoscale dispersion factors used for episodic releases are calculated differently and are based on the period of time modeled for dispersion of the release. Therefore, they have different units and give the total integrated air concentration directly without requiring division by the time period.

The decay-corrected source term, Q_e, and the episodic mesoscale dispersion factor, MDF_e, are used to calculate the total integrated air concentration according to the following equation:

$$TIC = Q_e \, MDF_e \qquad \text{(C-3)}$$

Substituting appropriate values from Table C-15 for IET #23 (FEET #2) in Equation (C-3) gives

$$TIC = (4.55 \; Ci) \left(3.37 \times 10^{-12} \; \frac{h}{m^3}\right)$$

$$TIC = 1.53 \times 10^{-11} \; \frac{Ci\text{-}h}{m^3}$$

The average air concentration for episodic releases depends on the TIC and the period of time over which the dispersion of the release is modeled and is given is given by the following equation:

$$AAC = \frac{TIC}{t_e} \qquad \text{(C-4)}$$

Substituting the calculated value of TIC and the modeled time for IET #23 (FEET #2) from Table C-15 into Equation (C-4) gives

$$AAC = \frac{1.53 \times 10^{-11} \; \frac{Ci\text{-}h}{m^3}}{310 \; h}$$

$$AAC = 4.95 \times 10^{-14} \; \frac{Ci}{m^3}$$

C.3.2.1 Air Immersion Dose

External radiation dose from immersion in contaminated air can be calculated from TIC using the following equation:

$$D_a = TIC\ DCF_a\ 1.00 \times 10^6\ 1.14 \times 10^{-4} \qquad \text{(C-5)}$$

Substituting the appropriate dose conversion factor for the infant thyroid dose from Table C-12 gives

$$D_{a\ Thyroid} = \left(1.53 \times 10^{-11}\ \frac{Ci\text{-}h}{m^3}\right)\left(2.07 \times 10^3\ \frac{mrem/y}{\mu Ci/m^3}\right) \times \left(1.0 \times 10^6\ \frac{\mu Ci}{Ci}\right)\left(1.14 \times 10^{-4}\ \frac{y}{h}\right)$$

$$D_{a\ Thyroid} = 3.62 \times 10^{-6}\ mrem$$

Substituting the appropriate dose conversion factor for the infant EDE from Table C-12 gives

$$D_{a\ EDE} = \left(1.53 \times 10^{-11}\ \frac{Ci\text{-}h}{m^3}\right)\left(1.91 \times 10^3\ \frac{mrem/y}{\mu Ci/m^3}\right) \times \left(1.0 \times 10^6\ \frac{\mu Ci}{Ci}\right)\left(1.14 \times 10^{-4}\ \frac{y}{h}\right)$$

$$D_{a\ EDE} = 3.34 \times 10^{-6}\ mrem$$

C.3.2.2 Inhalation Dose

Internal radiation dose from inhalation of contaminated air can be calculated from the TIC if the breathing rate is known by using the equation

$$D_i = TIC\ BR_j\ DCF_{ij}\ 1.00 \times 10^6\ 1.00 \times 10^3 \qquad \text{(C-6)}$$

Substituting the breathing rate and the appropriate dose conversion factor for the infant thyroid dose from Table C-12 gives

$$D_{i\ Thyroid} = \left(1.53 \times 10^{-11}\ \frac{Ci\text{-}h}{m^3}\right)\left(1.58 \times 10^{-1}\ \frac{m^3}{h}\right)\left(8.51\ \frac{rem}{\mu Ci}\right) \times \left(1.00 \times 10^{6}\ \frac{\mu Ci}{Ci}\right)\left(1.00 \times 10^{3}\ \frac{mrem}{rem}\right)$$

$$D_{i\ Thyroid} = 2.06 \times 10^{-2}\ mrem$$

Substituting the breathing rate and the appropriate dose conversion factor for the infant EDE from Table C-12 gives

$$D_{i\ EDE} = \left(1.53 \times 10^{-11}\ \frac{Ci\text{-}h}{m^3}\right)\left(1.58 \times 10^{-1}\ \frac{m^3}{h}\right)\left(2.55 \times 10^{-1}\ \frac{rem}{\mu Ci}\right) \times \left(1.00 \times 10^{6}\ \frac{\mu Ci}{Ci}\right)\left(1.00 \times 10^{3}\ \frac{mrem}{rem}\right)$$

$$D_{i\ EDE} = 6.18 \times 10^{-4}\ mrem$$

C.3.2.3 Ground Surface Dose

As with operational releases, calculation of external radiation dose from contaminated ground surfaces requires estimating radioactive decay over the period of deposition and during the 50-year dose commitment period following deposition. Dose from contaminated ground surfaces is given by

$$D_S = AAC\ V_d\ SF\ DCF_s\ \frac{1}{\lambda_r}\left(t - \frac{e^{-\lambda_r T}}{\lambda_r}(1 - e^{-\lambda_r t})\right) \times (3.60 \times 10^{3})\ (1.00 \times 10^{6})\ (1.14 \times 10^{-4}) \qquad \text{(C-7)}$$

To simplify this equation, the exponential terms will be evaluated first. For releases from IET #23 (FEET #2) the period of exposure, t, is 0.9 hour and the dose commitment period, T, is 50 years (4.38×10^5 h). Substituting in the bracketed expression from the equation above gives

$$t - \frac{e^{-\lambda_r T}}{\lambda_r}(1 - e^{-\lambda_r t}) =$$

$$(0.9h) - \frac{e^{-(3.59 \times 10^{-3}\ h^{-1})(4.38 \times 10^{5}\ h)}}{3.59 \times 10^{-3}\ h^{-1}}(1 - e^{-(3.59 \times 10^{-3}\ h^{-1})(310\ h)}) \approx$$

$$(0.9\ h) - \frac{0}{3.59 \times 10^{-3}\ h^{-1}}(1 - 9.97 \times 10^{-1}) = 310\ h$$

Substituting the simplified exponential term, appropriate parameter values from Table C-15, and the dose conversion factor for the infant thyroid dose from Table C-12 gives

$$D_{S\ Thyroid} = \left(4.95 \times 10^{-14}\frac{Ci}{m^3}\right)\left(0.018\frac{m}{s}\right)(0.7)\left(4.38 \times 10^{1}\ \frac{mrem/y}{\mu Ci/m^2}\right) \times$$

$$\left(\frac{310\ h}{3.59 \times 10^{-3} h^{-1}}\right)\left(3.60 \times 10^{3}\frac{s}{h}\right)\left(1.00 \times 10^{6}\frac{\mu Ci}{Ci}\right)\left(1.14 \times 10^{-4}\frac{y}{h}\right)$$

$$D_{S\ Thyroid} = 9.67 \times 10^{-4}\ mrem$$

Substituting the simplified exponential term, appropriate parameter values from Table C-15, and the dose conversion factor for the infant EDE from Table C-12 gives

$$D_{S\ EDE} = \left(4.95 \times 10^{-14}\frac{Ci}{m^3}\right)\left(0.018\frac{m}{s}\right)(0.7)\left(4.04 \times 10^{1}\ \frac{mrem/y}{\mu Ci/m^2}\right) \times$$

$$\left(\frac{310\ h}{3.59 \times 10^{-3} h^{-1}}\right)\left(3.60 \times 10^{3}\frac{s}{h}\right)\left(1.00 \times 10^{6}\frac{\mu Ci}{Ci}\right)\left(1.14 \times 10^{-4}\frac{y}{h}\right)$$

$$D_{S\ EDE} = 8.92 \times 10^{-4}\ mrem$$

C.3.2.4 Ingestion Dose

Calculating ingestion doses from episodic releases uses the same equation as for operational releases. However, the IUF differs because grazing beef and dairy cattle are conservatively assumed to be on pasture for the entire period over which dispersion of the release is modeled. For releases during the winter months of December, January, and February, the IUFs for short-lived radionuclides will also differ because of adjustments made to reflect reduced above ground plant biomass. The releases from IET #23 (FEET #2) occurred on October 26, 1960; therefore, no winter-time adjustments are necessary. Ingestion doses are given by

$$D_i = TIC\ V_d\ IUF\ DCF_i\ (1.00 \times 10^6)\ (3.60 \times 10^3)\ (1.00 \times 10^3) \qquad \text{(C-8)}$$

Substituting appropriate parameter values and the dose conversion factor for the infant thyroid dose from Table C-12 into Equation (C-8) gives

$$D_{i\ Thyroid} = \left(1.53 \times 10^{-11} \frac{Ci\text{-}h}{m^3}\right)\left(1.8 \times 10^{-2} \frac{m}{s}\right)\left(3.04 \times 10^{-1} \frac{\mu Ci}{\mu Ci/m^2}\right) \times$$

$$\left(1.37 \times 10^{+1} \frac{rem}{\mu Ci}\right)\left(1.00 \times 10^6 \frac{\mu Ci}{Ci}\right)\left(3.60 \times 10^3 \frac{s}{h}\right)\left(1.00 \times 10^3 \frac{mrem}{rem}\right)$$

$$D_{i\ Thyroid} = 4.14 \times 10^2\ mrem$$

Substituting appropriate parameter values and the dose conversion factor for the infant EDE from Table C-12 into Equation (C-8) gives

$$D_{i\ EDE} = \left(1.53 \times 10^{-11} \frac{Ci\text{-}h}{m^3}\right)\left(1.8 \times 10^{-2} \frac{m}{s}\right)\left(3.04 \times 10^{-1} \frac{\mu Ci}{\mu Ci/m^2}\right) \times$$

$$\left(4.07 \times 10^{-1} \frac{rem}{\mu Ci}\right)\left(1.00 \times 10^{6} \frac{\mu Ci}{Ci}\right)\left(3.60 \times 10^{3} \frac{s}{h}\right)\left(1.00 \times 10^{3} \frac{mrem}{rem}\right)$$

$$D_{i\ EDE} = 1.23 \times 10^{-1}\ mrem$$

C.3.2.5 Total Episodic Dose

The doses from all four pathways can now be added. Table C-16 presents the estimated thyroid dose and EDE for an infant at Monteview exposed to releases from IET #23 (FEET #2) on November 26, 1960. The totals for all four pathways are 4.16 mrem to the thyroid and 0.125 mrem EDE. As with the operational releases of I-131, the largest percentage of the dose results from ingestion.

In 1960, three episodic releases (IETs #18, #19, and #25) overlapped the winter months of January, February, and December. Doses for the four pathways were calculated in the same manner as demonstrated for IET #23 (FEET #2), except for the ingestion dose. During the winter months, the IUF was reduced to 10% of the summer value shown in Table C-11 for radionuclides with half-lives less than 10 days.

Table C-16. Summary of I-131 doses from IET #23 (FEET #2) releases for an infant residing at "Building"

Pathway	EDE (mrem)	Thyroid (mrem)
Air immersion	3.62E-06	3.62E-06
Inhalation	6.18E-04	2.06E-02
Ground surface	8.92E-04	9.67E-04
Ingestion	1.23E-01	4.14E+00
Total	1.25E-01	4.16E+00

C.3.3 1960 Combined Operational and Episodic Releases

Combining operational and episodic doses requires taking into account doses from each source at locations other than those given by the maximum operational and episodic MDFs. In principle, it is possible operational and episodic releases to combine to give in a larger total dose at a third location that is not the maximum for either the operational or episodic release. In order to illustrate this, the I-131 doses calculated in the preceding sections will be evaluated.

Radiation dose from all pathways is proportional to the total integrated air concentration, which is proportional to the MDF. Therefore, dose at locations other than that for the maximum MDF may be calculated if the corresponding MDF for the location of interest is known. The procedure is similar for both operational and episodic releases and uses the following equation:

$$EDE_{Loc} = EDE_{Max}\left(\frac{MDF_{Loc}}{MDF_{Max}}\right)$$

The parameters needed for calculating combined doses from operational and episodic releases of I-131 during 1960 are presented in Table C-17. The last two columns of this table indicates the report section or table from which the value was taken. Substituting appropriate values of EDE for operational and episodic releases and values for MDFs from Table C-15 gives the values presented in Table C-18. These illustrate that IET #23 (FEET #2) resulted in a significant contribution to the total EDE for infants at both Monteview and Atomic City.

This example has included only I-131 and evaluated only doses to infants. However, the total radiation dose from all radionuclides released during 1960 shows a pattern similar to the values presented in Table C-16. In this case, the location of the maximally exposed individual for 1960 would be reported as Atomic City.

Table C-17. Operational and episodic release mesoscale dispersion factors for 1960

Parameter[a]	Value	Units	Source of Data: Section Number	Source of Data: Table Number
Operational releases, 1960				
Atomic City[a]	5.00E-08	h^2/m^3	B.4	B-1
Monteview	2.60E-08	h^2/m^3	not shown, see Figure B-4	
Episodic release, IET #23 (FEET #2)				
Atomic City	1.68E-12	h/m^3	not shown	
Monteview[b]	3.37E-12	h/m^3	B.4	B-2

a. Maximum MDF value for operational releases in 1960.

b. Maximum MDF value for IET #23 (FEET #2).

Table C-18. Combined EDE from IET #23 (FEET #2) and operational releases during 1960 (mrem)[a]

Location	Operational	Episodic	Total
Atomic City	0.618	0.062	0.680
Monteview	0.247	0.125	0.372

a. Doses are for I-131 in infants only.

C.4 RESULTS OF DOSE CALCULATIONS

The complete results of the dose calculations have been converted to microfiche and are included in the back of this volume of the report. The results were too voluminous to print. The operational release dose results are summarized in Table C-19. The episodic release dose results are summarized in Table C-20. The operational and episodic doses were summed for each year as outlined in Section C.3.3. The results of those summations are shown in Table C-21.

The results of the dose calculations are plotted for operational releases for an adult in Figure C-1. The EDE is plotted as a solid line, and the maximum organ dose is plotted as a dashed line. The EDE is the sum of the dose from external sources of radiation and the committed effective dose equivalent (CEDE) from internally deposited radionuclides. The CEDE is the sum of the products of the committed dose equivalents to organs or tissues of the body multiplied by the appropriate weighting factor. The weighting factors used each reflect the relative carcinogenic or genetic risk resulting from exposure of the individual organ to radioactivity. The EDE is, therefore, proportional to the risk of detrimental health effects occurring from exposure to radioactive material, and it provides a common basis for comparing the relative importance of different episodic events and operational releases.

The estimated EDEs to an adult from operational releases range over three orders of magnitude from 11 mrem in 1957 to 0.006 mrem in 1989 (see Figure C-1). The decreasing trend in EDE since 1957 is due to the lower airborne releases of radioactivity as a result of changes in facility missions, completion of reactor testing programs involving the Materials Testing Reactor (MTR), improvements in process design, and improvements in emission control systems. The highest EDEs occurred from 1957 through 1959 and were primarily due to the processing of reactor fuels for RaLa at the Idaho Chemical Processing Plant (ICPP) (see Section A.2.2.4). In 1957, 64% of the EDE was due to I-131, 17% was due to Cs-137, and 11% was due to Sr-90. Improved controls reduced the emissions of radioiodines from the RaLa project. By 1959, the I-131 contribution to the EDE declined to 19%, while Cs-137 was

Table C-19. Operational dose summary[a]

Year	Effective Dose Equivalent (mrem) Adult	Child	Infant	Radiation Protection Standard[b] (mrem/yr)	Maximum Organ Dose (mrem) Adult	Organ	Child	Organ	Infant	Organ	Radiation Protection Standard (mrem/yr)
1989	6.19E-03	6.33E-03	6.03E-03	25	4.43E-02	Skin	4.46E-02	Skin	4.44E-02	Skin	75
1988[c]	1.47E-01	1.65E-01	1.37E-01	25	2.00E+00	Thyroid	2.66E+00	Thyroid	1.82E+00	Thyroid	75
1987	1.10E-01	1.17E-01	1.04E-01	25	9.31E-01	Thyroid	1.22E+00	Thyroid	8.50E-01	Thyroid	75
1986	2.97E-02	3.04E-02	2.85E-02	25	1.06E-01	Thyroid	1.33E-01	Thyroid	9.97E-02	Thyroid	75
1985	2.43E-02	2.51E-02	2.34E-02	25	1.79E-01	Skin	1.80E-01	Skin	1.79E-01	Skin	75
1984	2.65E-02	3.08E-02	2.38E-02	500	5.32E-01	Thyroid	7.15E-01	Thyroid	4.84E-01	Thyroid	1500
1983	4.39E-02	4.80E-02	3.77E-02	500	6.88E-01	Thyroid	9.21E-01	Thyroid	6.25E-01	Thyroid	1500
1982	3.37E-02	3.68E-02	3.18E-02	500	3.95E-01	Thyroid	5.24E-01	Thyroid	3.60E-01	Thyroid	1500
1981	3.97E-02	4.24E-02	3.80E-02	500	3.05E-01	Thyroid	4.04E-01	Thyroid	3.12E-01	Thyroid	1500
1980	7.90E-02	8.43E-02	7.02E-02	500	6.47E-01	Thyroid	8.53E-01	Thyroid	6.01E-01	Thyroid	1500
1979	7.19E-02	7.37E-02	6.74E-02	500	4.82E-01	Thyroid	6.31E-01	Thyroid	4.42E-01	Thyroid	1500
1978	7.90E-02	8.54E-02	6.88E-02	500	7.85E-01	Thyroid	1.04E+00	Thyroid	7.23E-01	Thyroid	1500
1977[d]	1.27E-01	1.31E-01	1.13E-01	500	9.54E-01	Thyroid	1.25E+00	Thyroid	8.69E-01	Thyroid	1500
1976	2.92E-01	2.95E-01	2.78E-01	500	1.29E+00	Thyroid	1.64E+00	Thyroid	1.19E+00	Thyroid	1500
1975	1.70E-01	1.68E-01	1.62E-01	500	6.46E-01	Skin	6.54E-01	Thyroid	6.51E-01	Skin	1500
1974	2.54E-01	2.39E-01	2.36E-01	500	1.70E+00	Skin	1.72E+00	Skin	1.72E+00	Skin	1500
1973[c]	3.08E-01	2.90E-01	2.85E-01	500	1.77E+00	Skin	1.78E+00	Skin	1.79E+00	Skin	1500
1972	5.06E-01	4.92E-01	5.27E-01	500	4.10E+00	Skin	4.13E+00	Skin	4.13E+00	Skin	1500
1971	1.82E+00	1.78E+00	1.87E+00	500	1.73E+01	Skin	1.74E+01	Skin	1.74E+01	Skin	1500
1970	4.42E-01	4.24E-01	4.17E-01	500	3.90E+00	Skin	3.92E+00	Skin	3.92E+00	Skin	1500
1969	6.57E-01	6.48E-01	7.15E-01	500	5.01E+00	Skin	5.04E+00	Skin	5.04E+00	Skin	1500
1968	1.42E+00	1.40E+00	1.44E+00	500	1.63E+01	Skin	1.64E+01	Skin	1.64E+01	Skin	1500
1967	7.54E-01	7.89E-01	7.82E-01	500	4.52E+00	Skin	4.53E+00	Skin	4.53E+00	Skin	1500
1966	2.13E+00	2.33E+00	2.31E+00	500	2.04E+01	Skin	2.05E+01	Skin	2.05E+01	Skin	1500
1965	3.55E+00	3.41E+00	3.55E+00	500	3.51E+01	Skin	3.52E+01	Skin	3.52E+01	Skin	1500
1964	3.19E+00	3.65E+00	3.50E+00	500	3.14E+01	Skin	3.14E+01	Skin	3.15E+01	Skin	1500
1963	4.52E+00	4.37E+00	4.72E+00	500	3.69E+01	Skin	3.71E+01	Skin	3.71E+01	Skin	1500
1962	1.35E+00	1.43E+00	1.89E+00	500	7.78E+00	Thyroid	1.19E+01	Thyroid	2.68E+01	Thyroid	1500
1961	1.27E+00	1.37E+00	1.83E+00	500	8.05E+00	Thyroid	1.24E+01	Thyroid	2.79E+01	Thyroid	1500
1960	1.02E+00	1.05E+00	1.38E+00	500	6.08E+00	Thyroid	9.37E+00	Thyroid	2.12E+01	Thyroid	1500
1959	5.92E+00	6.22E+00	8.71E+00	500	4.47E+01	Skin	6.27E+01	Thyroid	1.44E+02	Thyroid	1500
1958	1.09E+01	1.34E+01	2.47E+01	500	1.73E+02	Thyroid	2.77E+02	Thyroid	6.51E+02	Thyroid	1500
1957	1.12E+01	1.49E+01	3.03E+01	500	2.33E+02	Thyroid	3.74E+02	Thyroid	8.82E+02	Thyroid	1500
1956	3.34E+00	3.45E+00	3.52E+00	1500	3.35E+01	Skin	3.36E+01	Skin	3.36E+01	Skin	2500
1955	3.42E+00	3.49E+00	3.46E+00	1500	3.19E+01	Skin	3.20E+01	Skin	3.21E+01	Skin	2500
1954	2.42E+00	2.46E+00	2.43E+00	1500	2.25E+01	Skin	2.26E+01	Skin	2.26E+01	Skin	2500
1953	1.64E+00	1.64E+00	1.64E+00	1500	1.22E+01	Skin	1.23E+01	Skin	1.23E+01	Skin	2500
1952	4.33E-01	4.25E-01	4.32E-01	1500	2.09E+00	Skin	2.09E+00	Skin	2.09E+00	Skin	2500

a. Doses are calculated for location of highest annual average atmospheric concentration. This corresponds to Atomic City unless otherwise indicated.

Table C-19. (continued)

b. The radiation protection standard shown is that for the whole body and is not strictly comparable to the calculated EDE. The EDE provides a common basis for comparing the importance of various releases.

c. Doses are calculated at a point near Terreton during 1973 and 1988.

d. Doses are calculated at a point about 8 mi WSW of Hamer during 1977.

Table C-20. Episodic dose summary

Dates	Episodes	Effective Dose Equivalent (mrem)			Maximum Organ Dose (mrem)					
		Adult	Child	Infant	Adult	Organ	Child	Organ	Infant	Organ
9/22/67	CERT No. 22	5.43E-03	1.01E-02	2.91E-02	1.78E-01	Thyroid	3.40E-01	Thyroid	9.73E-01	Thyroid
7/6/67	CERT No. 20	9.51E-04	1.77E-03	5.10E-03	3.11E-02	Thyroid	5.96E-02	Thyroid	1.70E-01	Thyroid
7/21/66	CERT No. 11	3.28E-04	5.32E-04	8.71E-04	1.10E-02	Thyroid	1.75E-02	Thyroid	2.90E-02	Thyroid
6/14/66	CERT No. 10	1.99E-03	3.70E-03	1.07E-02	6.51E-02	Thyroid	1.25E-01	Thyroid	3.57E-01	Thyroid
1/11/66	SNAPTRAN-2	2.20E-02	1.86E-02	1.36E-02	8.91E-02	Lungs	6.16E-02	Lungs	3.15E-02	Skin
11/22/65	CERT No. 7	1.05E-02	1.96E-02	5.65E-02	3.45E-01	Thyroid	6.61E-01	Thyroid	1.89E+00	Thyroid
9/14/65	CERT No. 6	1.80E-04	2.92E-04	4.78E-04	6.07E-03	Thyroid	9.62E-03	Thyroid	1.59E-02	Thyroid
6/10/65	CERT No. 5	6.63E-04	1.23E-03	3.55E-03	2.17E-02	Thyroid	4.16E-02	Thyroid	1.19E-01	Thyroid
5/27/65	CERT No. 4	1.77E-06	3.29E-06	9.49E-06	5.79E-05	Thyroid	1.11E-04	Thyroid	3.17E-04	Thyroid
12/11/64	CERT No. 3	4.40E-03	7.13E-03	1.71E-02	1.13E-01	Thyroid	2.07E-01	Thyroid	5.40E-01	Thyroid
9/2/64	CERT No. 2	6.27E-03	1.16E-02	3.36E-02	2.05E-01	Thyroid	3.93E-01	Thyroid	1.12E+00	Thyroid
4/14/64	SPERT-I, Test No. 3	5.25E-06	4.98E-06	4.66E-06	1.53E-05	Skin	1.53E-05	Skin	1.53E-05	Skin
4/1/64	SNAPTRAN-3	1.35E-02	1.35E-02	1.35E-02	2.50E-02	Skin	2.50E-02	Skin	2.50E-02	Skin
11/10/63	SPERT-I, Test No. 2	6.61E-09	6.05E-09	5.42E-09	2.32E-08	Skin	2.32E-08	Skin	2.32E-08	Skin
5/27/63	CERT No. 1	7.54E-03	1.40E-02	4.04E-02	2.46E-01	Thyroid	4.72E-01	Thyroid	1.35E+00	Thyroid
11/5/62	SPERT-I, Test No. 1	4.73E-05	4.42E-05	4.03E-05	1.39E-04	Skin	1.39E-04	Skin	1.39E-04	Skin
1/25/61	1961 ICPP Criticality	2.75E-03	2.67E-03	2.91E-03	1.39E-02	Thyroid	1.67E-02	Thyroid	3.12E-02	Thyroid
1/3/61	SL-1 Accident	3.18E+00	4.21E+00	7.45E+00	3.94E+01	Thyroid	7.10E+01	Thyroid	1.83E+02	Thyroid
3/01-30/61	IET #26(B)	3.57E-02	5.69E-02	1.45E-01	8.69E-01	Thyroid	1.61E+00	Thyroid	4.53E+00	Thyroid
12/23-2/28/61	IET #26(A)	4.74E-02	5.16E-02	6.78E-02	2.66E-01	Thyroid	4.12E-01	Thyroid	9.54E-01	Thyroid
12/1-15/60	IET #25(B)	3.04E-02	3.30E-02	4.30E-02	1.77E-01	Thyroid	2.65E-01	Thyroid	6.00E-01	Thyroid
11/22-30/60	IET #25(A)	2.58E-02	3.35E-02	6.63E-02	3.72E-01	Thyroid	6.43E-01	Thyroid	1.73E+00	Thyroid
10/26/60	IET #24 (LIME-II)	5.10E-02	5.67E-02	8.31E-02	3.92E-01	Thyroid	6.03E-01	Thyroid	1.49E+00	Thyroid
9/7-10/14/60	IET #23 (FEET #2)	3.22E-02	5.17E-02	1.33E-01	8.21E-01	Thyroid	1.50E+00	Thyroid	4.19E+00	Thyroid
8/25/60	IET #22 (LIME)	1.01E-02	1.28E-02	2.41E-02	1.22E-01	Thyroid	2.18E-01	Thyroid	5.99E-01	Thyroid
6/29-8/6/60	IET #21 (FEET #1)	1.63E-02	2.36E-02	5.43E-02	3.13E-01	Thyroid	5.73E-01	Thyroid	1.59E+00	Thyroid
5/14-6/10/60	IET #20	3.65E-02	5.52E-02	1.33E-01	8.09E-01	Thyroid	1.47E+00	Thyroid	4.07E+00	Thyroid
3/1-4/30/60	IET #19(B)	7.93E-02	1.23E-01	3.06E-01	1.85E+00	Thyroid	3.39E+00	Thyroid	9.46E+00	Thyroid
2/17-2/29/60	IET #19(A)	1.80E-02	2.02E-02	2.90E-02	1.50E-01	Thyroid	2.26E-01	Thyroid	5.19E-01	Thyroid
1/6-2/7/60	IET #18	9.93E-02	1.15E-01	1.75E-01	9.40E-01	Thyroid	1.48E+00	Thyroid	3.49E+00	Thyroid
12/1-12/12/59	IET #17(B)	1.47E-02	1.61E-02	2.12E-02	8.39E-02	Thyroid	1.30E-01	Thyroid	3.02E-01	Thyroid
11/2-30/59	IET #17(A)	3.43E-02	4.50E-02	8.96E-02	4.70E-01	Thyroid	8.46E-01	Thyroid	2.33E+00	Thyroid
10/16/59	1959 ICCP Criticality	1.12E+00	1.17E+00	1.52E+00	5.94E+00	Thyroid	9.10E+00	Thyroid	2.23E+01	Thyroid
10/9/59	IET #16	1.11E-03	1.12E-03	1.17E-03	2.16E-03	Skin	2.17E-03	Thyroid	3.75E-03	Thyroid
6/16-24/59	IET #15(B)	1.60E-02	2.60E-02	6.78E-02	4.20E-01	Thyroid	7.73E-01	Thyroid	2.16E+00	Thyroid
6/3-15/59	IET #15(A)	2.05E-03	2.41E-03	3.93E-03	1.76E-02	Thyroid	3.03E-02	Thyroid	8.08E-02	Thyroid
4/24-5/19/59	IET #14	7.79E-02	1.24E-01	3.16E-01	1.97E+00	Thyroid	3.59E+00	Thyroid	9.97E+00	Thyroid
11/18/58	IET #13 (HTRE No. 3 Excursion)	1.20E-01	1.20E-01	1.56E-01	7.77E-01	Thyroid	1.21E+00	Thyroid	3.02E+00	Thyroid
10/29-30/58	FECF Filter Break	1.14E-01	1.21E-01	1.20E-01	1.35E+00	Skin	1.35E+00	Skin	1.36E+00	Skin
9/26/58	FPFRT-9	1.55E+00	1.46E+00	1.50E+00	1.19E+01	Skin	1.20E+01	Skin	1.20E+01	Skin
9/18/58	FPFRT-8	1.36E+01	1.45E+01	1.71E+01	6.61E+01	Skin	7.24E+01	Thyroid	1.89E+02	Thyroid
9/17/58	FPFRT-7	9.48E-01	1.01E+00	1.25E+00	4.27E+00	Skin	5.80E+00	Thyroid	1.53E+01	Thyroid
9/4/58	FPFRT-6	1.91E-02	2.24E-02	3.51E-02	1.40E-01	Thyroid	2.58E-01	Thyroid	7.14E-01	Thyroid
8/27/58	FPFRT-5	1.66E+00	2.11E+00	3.90E+00	1.84E+01	Thyroid	3.43E+01	Thyroid	9.65E+01	Thyroid

Table C-20. (continued)

Dates	Episodes	Effective Dose Equivalent (mrem)			Maximum Organ Dose (mrem)					
		Adult	Child	Infant	Adult	Organ	Child	Organ	Infant	Organ
8/14/58	FPFRT-4	4.05E+00	3.80E+00	3.92E+00	3.20E+01	Skin	3.23E+01	Skin	3.23E+01	Skin
8/6/58	FPFRT-3	3.78E+00	3.54E+00	3.65E+00	2.90E+01	Skin	2.92E+01	Skin	2.93E+01	Skin
8/4/58	FPFRT-2	1.30E-01	1.22E-01	1.26E-01	1.00E+00	Skin	1.01E+00	Skin	1.01E+00	Skin
7/25/58	FPFRT-1	8.62E-02	8.09E-02	8.34E-02	6.63E-01	Skin	6.68E-01	Skin	6.69E-01	Skin
5/2/58	IET #12 (BOOT)	1.39E-02	2.09E-02	5.13E-02	3.17E-01	Thyroid	5.76E-01	Thyroid	1.60E+00	Thyroid
3/20-4/14/58	IET #11	3.34E-02	5.35E-02	1.37E-01	8.65E-01	Thyroid	1.57E+00	Thyroid	4.36E+00	Thyroid
3/1-6/58	IET #10(B)	4.32E-01	5.00E-01	7.92E-01	3.32E+00	Thyroid	5.81E+00	Thyroid	1.57E+01	Thyroid
12/20/57-2/25/58	IET #10(A)	6.68E-01	6.86E-01	7.68E-01	2.17E+00	Thyroid	2.92E+00	Thyroid	5.92E+00	Thyroid
7/31-8/28/57	IET #8	7.15E-03	5.84E-03	4.36E-03	1.79E-02	Skin	1.79E-02	Skin	1.79E-02	Skin
3/20/57	Fuel Element Burn Test, B	5.31E+00	5.70E+00	5.61E+00	6.05E+01	Skin	6.07E+01	Skin	6.07E+01	Skin
3/20/57	Fuel Element Burn Test, A	1.07E-03	1.49E-03	3.02E-03	1.52E-02	Thyroid	2.87E-02	Thyroid	8.15E-02	Thyroid
12/18/56	IET #6	1.90E-03	2.01E-03	2.52E-03	1.11E-02	Thyroid	1.66E-02	Thyroid	3.75E-02	Thyroid
6/29/56	IET #4(C)	4.12E+00	5.77E+00	1.28E+01	7.47E+01	Thyroid	1.35E+02	Thyroid	3.74E+02	Thyroid
5/24-6/26/56	IET #4(B)	1.60E+00	2.05E+00	4.09E+00	2.79E+01	Thyroid	4.39E+01	Thyroid	1.12E+02	Thyroid
5/1-23/56	IET #4(A)	7.02E-02	9.09E-02	1.84E-01	1.26E+00	Thyroid	2.00E+00	Thyroid	5.10E+00	Thyroid
2/11-24/56	IET #3	2.88E+01	3.39E+01	5.44E+01	3.22E+02	Thyroid	5.14E+02	Thyroid	1.23E+03	Thyroid
6/18-7/1/55	NRF S1W Engineering Test	1.13E-01	1.98E-01	5.48E-01	3.48E+00	Thyroid	6.44E+00	Thyroid	1.81E+01	Thyroid
7/22/54	BORAX-I Excursion	7.21E-03	6.82E-03	6.43E-03	2.12E-02	Skin	2.12E-02	Skin	2.12E-02	Skin

Table C-21. Total annual dose summary

Year	Location	Effective Dose Equivalent (mrem) Adult	Child	Infant	Radiation Protection Standard[a] (mrem/yr)	Maximum Organ Dose (mrem) Adult	Organ	Child	Organ	Infant	Organ	Radiation Protection Standard (mrem/yr)
1989	Atomic City	6.19E-03	6.33E-03	6.03E-03	25	4.43E-02	Skin	4.46E-02	Skin	4.44E-02	Skin	75
1988	Terreton	1.47E-01	1.65E-01	1.37E-01	25	2.00E+00	Thyroid	2.66E+00	Thyroid	1.82E+00	Thyroid	75
1987	Atomic City	1.10E-01	1.17E-01	1.04E-01	25	9.31E-01	Thyroid	1.22E+00	Thyroid	8.50E-01	Thyroid	75
1986	Atomic City	2.97E-02	3.04E-02	2.85E-02	25	1.06E-01	Thyroid	1.33E-01	Thyroid	9.97E-02	Thyroid	75
1985	Atomic City	2.43E-02	2.51E-02	2.34E-02	25	1.79E-01	Skin	1.80E-01	Skin	1.79E-01	Skin	75
1984	Atomic City	2.65E-02	3.08E-02	2.38E-02	500	5.32E-01	Thyroid	7.15E-01	Thyroid	4.84E-01	Thyroid	1500
1983	Atomic City	4.39E-02	4.80E-02	3.77E-02	500	6.88E-01	Thyroid	9.21E-01	Thyroid	6.25E-01	Thyroid	1500
1982	Atomic City	3.37E-02	3.68E-02	3.18E-02	500	3.95E-01	Thyroid	5.24E-01	Thyroid	3.60E-01	Thyroid	1500
1981	Atomic City	3.97E-02	4.24E-02	3.80E-02	500	3.05E-01	Thyroid	4.04E-01	Thyroid	3.12E-01	Thyroid	1500
1980	Atomic City	7.90E-02	8.43E-02	7.02E-02	500	6.47E-01	Thyroid	8.53E-01	Thyroid	6.01E-01	Thyroid	1500
1979	Atomic City	7.19E-02	7.37E-02	6.74E-02	500	4.82E-01	Thyroid	6.31E-01	Thyroid	4.42E-01	Thyroid	1500
1978	Atomic City	7.90E-02	8.54E-02	6.88E-02	500	7.85E-01	Thyroid	1.04E+00	Thyroid	7.23E-01	Thyroid	1500
1977	Hamer	1.27E-01	1.31E-01	1.13E-01	500	9.54E-01	Thyroid	1.25E+00	Thyroid	8.69E-01	Thyroid	1500
1976	Atomic City	2.92E-01	2.95E-01	2.78E-01	500	1.29E+00	Thyroid	1.64E+00	Thyroid	1.19E+00	Thyroid	1500
1975	Atomic City	1.70E-01	1.68E-01	1.62E-01	500	6.46E-01	Skin	6.54E-01	Thyroid	6.51E-01	Skin	1500
1974	Atomic City	2.54E-01	2.39E-01	2.36E-01	500	1.70E+00	Skin	1.72E+00	Skin	1.72E+00	Skin	1500
1973	Terreton	3.08E-01	2.90E-01	2.85E-01	500	1.77E+00	Skin	1.78E+00	Skin	1.79E+00	Skin	1500
1972	Atomic City	5.06E-01	4.92E-01	5.27E-01	500	4.10E+00	Skin	4.13E+00	Skin	4.13E+00	Skin	1500
1971	Atomic City	1.82E+00	1.78E+00	1.87E+00	500	1.73E+01	Skin	1.74E+01	Skin	1.74E+01	Skin	1500
1970	Atomic City	4.42E-01	4.24E-01	4.17E-01	500	3.90E+00	Skin	3.92E+00	Skin	3.92E+00	Skin	1500
1969	Atomic City	6.57E-01	6.48E-01	7.15E-01	500	5.01E+00	Skin	5.04E+00	Skin	5.04E+00	Skin	1500
1968	Atomic City	1.42E+00	1.40E+00	1.44E+00	500	1.63E+01	Skin	1.64E+01	Skin	1.64E+01	Skin	1500
1967	Atomic City	7.54E-01	7.89E-01	7.82E-01	500	4.52E+00	Skin	4.53E+00	Skin	4.53E+00	Skin	1500
1966	Atomic City	2.13E+00	2.33E+00	2.31E+00	500	2.04E+01	Skin	2.05E+01	Skin	2.05E+01	Skin	1500
1965	Frenchman's Cabin	3.55E+00	3.42E+00	3.55E+00	500	3.51E+01	Skin	3.52E+01	Skin	3.52E+01	Skin	1500
1964	Atomic City	3.19E+00	3.65E+00	3.50E+00	500	3.14E+01	Skin	3.14E+01	Skin	3.15E+01	Skin	1500
1963	Atomic City	4.52E+00	4.37E+00	4.72E+00	500	3.69E+01	Skin	3.71E+01	Skin	3.71E+01	Skin	1500
1962	Atomic City	1.35E+00	1.43E+00	1.89E+00	500	7.78E+00	Thyroid	1.19E+01	Thyroid	2.68E+01	Thyroid	1500
1961[b]	Atomic City	4.46E+00	5.58E+00	9.28E+00	500	4.76E+01	Thyroid	8.36E+01	Thyroid	2.11E+02	Thyroid	1500
1960	Atomic City	1.22E+00	1.30E+00	1.84E+00	500	8.63E+00	Thyroid	1.37E+01	Thyroid	3.24E+00	Thyroid	3000
1959	Frenchman's Cabin	7.15E+00	7.57E+00	1.07E+00	500	4.87E+00	Thyroid	7.65E+01	Thyroid	1.79E+02	Thyroid	3000
1958	Mud Lake	2.86E+01	3.16E+01	4.47E+01	500	1.93E+02	Thyroid	3.17E+02	Thyroid	7.79E+02	Thyroid	3000
1957	Atomic City	1.12E+01	1.49E+01	3.03E+01	500	2.33E+02	Thyroid	3.74E+02	Thyroid	8.82E+02	Thyroid	1500
1956	Building	3.25E+01	3.81E+01	6.08E+01	1500	3.54E+02	Thyroid	5.63E+02	Thyroid	1.35E+03	Thyroid	1500
1955	Frenchman's Cabin	3.52E+00	3.66E+00	3.93E+00	1500	3.20E+01	Skin	3.21E+01	Skin	3.21E+01	Skin	2500
1954	Frenchman's Cabin	2.42E+00	2.46E+00	2.44E+00	1500	2.25E+01	Skin	2.26E+01	Skin	2.27E+01	Skin	2500
1953	Atomic City	1.64E+00	1.64E+00	1.64E+00	1500	1.22E+01	Skin	1.23E+01	Skin	1.23E+01	Skin	2500
1952	Atomic City	4.33E-01	4.25E-01	4.32E-01	1500	2.09E+00	Skin	2.09E+00	Skin	2.09E+00	Skin	2500

a. The radiation protection standard shown is that for the whole body and is not strictly comparable to the calculated EDE. The EDE provides a common basis for comparing the importance of various releases.
b. Before 1961, the estimates of total dose are all included in the operational release estimates. (see Table 4-1).

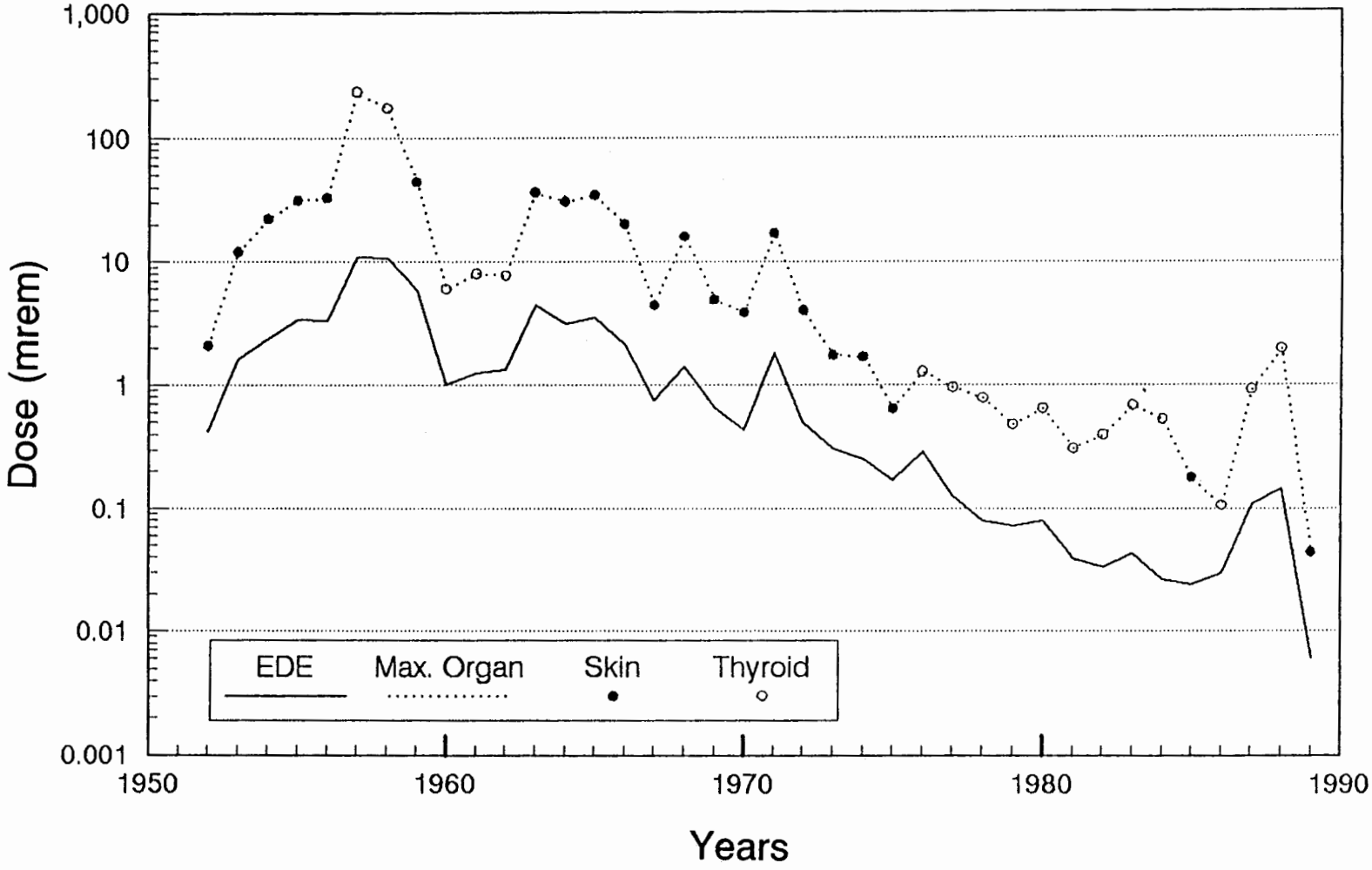

Figure C-1. Maximally exposed adult operational dose estimates.

38% of the total. From 1960 to 1962, the effluents from the TRA, particularly the MTR reactor, make a substantial contribution to the estimated EDE. Argon-41 accounts for 32% of the EDE in 1962. Krypton-88, Sr-90, I-131, and Cs-137 are also important contributors to the EDE during 1962. Cesium-137 was the highest dose contributor to the EDE in 11 of the 21 years from 1954 to 1974. It accounts for a large fraction of the peak EDEs in 1963, 1965, and 1971. Ruthenium-106 was a large contributor in 1964 and 1966. The peak in 1968 was largely due to Sr-90 and Cs-137.

Some variations occur in the calculated doses for operational releases (Figure C-1) from year to year because of intermittent facility operations or campaigns. For example, during 1989, no fuel was processed in the FAST facility at the ICPP and the NWCF did not operate during most of the year. The ATR reactor operated only about 50% of the time during 1989 because of a console upgrade. During 1987 and 1988, the EDE was higher than in the years before and after. This was caused by releases of Sb-125 during fuel processing at the FAST facility at the ICPP (see Section E.5.1) in addition to the releases of I-129. The FAST facility began processing fuel in the last quarter of 1986, and the campaign continued through the first half of 1988. Important processes at the ICPP were not in operation for most of 1985 and 1986. The ATR reactor only operated during 60% of 1986; therefore, the resulting doses for 1985 and 1986 are lower.

The maximum organ dose equivalent to an adult from operational releases (see Figure C-1) follows the same trend as the EDE, increasing from 1952 through 1957, and then generally decreasing through 1989. The largest organ dose equivalent ranges from 230 mrem in 1957 to 0.04 mrem in 1989. In the years between 1952 and 1956 and 1963 through 1975, the dose equivalent to the skin is higher than the dose to the thyroid and other organs. For these years, external exposure pathways from air immersion and ground deposition usually dominate, and the major contributors to the skin dose are Sr-90; Cs-137; Ce-144; and (in 1964, 1966, and 1971) Ru-106. Noble gases, such as Ar-41, Kr-88, and Kr-85, are also contributors to the skin dose during these years. From 1957 through 1962 (except in 1959), the largest organ dose was received by the thyroid, primarily as a result of I-131 releases from RaLa. For most of the years from 1976 to 1989, the thyroid also has the largest dose

equivalent. However, for this period of time the primary contributor to the thyroid dose is long-lived I-129 from fuel processing and waste calcination, rather than the I-131 characteristic of the 1950s. A relatively large quantity of I-129 was released in 1964 from the Waste Calcining Facility, but the thyroid dose in that year was exceeded by the skin dose because of particulate emissions. In August 1975, new emission control filters, referred to as the Atmospheric Protection System, were installed on the main stack at the ICPP. This system reduced particulate emissions, including Sr-90, Cs-137, Ce-144, and Ru-106, but it did not decrease the gaseous emissions of the various chemical forms of I-129. This caused the shift from the skin to the thyroid as the organ with the largest annual dose.

Figure C-2 shows the EDE and maximum organ doses for the infant from operational releases. As with the adult, the EDE and the maximum organ doses tend to increase from 1952 through 1957 and then generally decrease from 1957 to 1989. The infant EDE ranges from 30 mrem in 1957 to 0.006 mrem in 1989, while the largest organ dose equivalent ranges from 880 mrem in 1957 to 0.04 mrem in 1989. In general, the same radionuclides that produce the majority of the dose to an adult also produce the majority of the dose to the infant. For those years in which I-131 releases from RaLa were dominant (1957 through 1962), the thyroid doses are higher for the infant than for the adult because of the smaller thyroid and a larger fractional iodine uptake than adults, which results in a larger ingestion dose conversion factor for I-131. The thyroid doses calculated for an infant assume that an infant is fed milk from cows grazing near the INEL Site boundary. No reductions In I-131 intake are assumed for infants consuming milk from other sources such as breast milk, formula, or milk collected regionally by commercial dairies.

The dose estimates for a child were not plotted because they are similar to those received by an adult. One variation in the doses between the adult and the child results from age-dependent differences in the ingestion dose conversion factors for I-129. The dose equivalent to the thyroid of a child from ingestion of I-129 exceeds the dose equivalent calculated for the adult thyroid. The dose to a child's thyroid is about 35% higher than the values shown in Figure C-1 for the adult thyroid from 1976 through 1988.

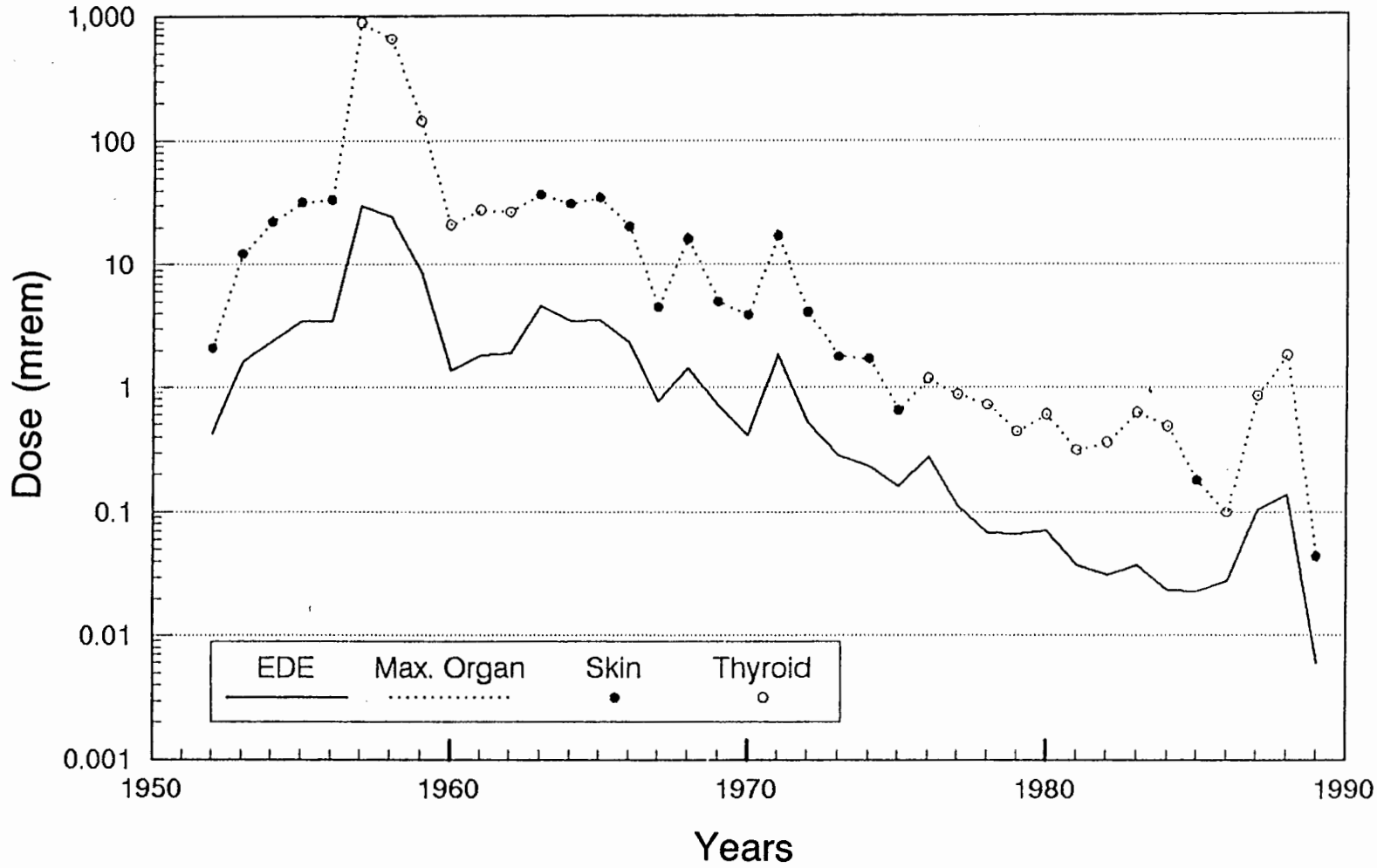

Figure C-2. Maximally exposed infant operational dose estimates.

Figure C-3 presents an overlay of the information contained in Figures C-1 and C-2. The EDEs and largest organ doses for the infant and adult are quite similar, except for 1957 to 1962. The difference from 1957 through 1962 is primarily due to the I-131 releases during RaLa resulting in higher thyroid dose equivalent and EDE for an infant compared to an adult.

The episodic release dose results were shown in Table C-20. The highest EDEs for episodic releases were for IET #3, Fuel Element Burn Test B, FPFRT-8, IET #4(C), and the SL-1 Accident. Most of the other tests with an EDE above 0.1 mrem are in the IET and FPFRT series. Only three of the episodic releases above 0.1 mrem EDE are accidents: the SL-1 Accident, the 1959 ICPP Criticality, and the FECF Filter Break. The EDEs for the episodic releases range from 54 mrem (infant) for IET #3 to less than 0.001 mrem for some of the tests. The maximum organ doses range from 1230 mrem to less than 0.001 mrem. Most of the maximum organ doses are to the thyroid; in those cases, the infant is the age group with the highest organ dose. The skin is the maximum organ for 13 episodic releases, and the lung is the maximum organ only for the adult and child for SNAPTRAN-2. The most important radionuclide for IET #3 and the SL-1 Accident was I-131. For the FPFRT release, the highest EDE comes from Zr-95, while the highest skin dose is from Sr-90. Cesium-137, Ce-144, Y-91, and Sr-89 are also important. For Fuel Element Burn Test B, the highest EDE is due to Cs-137, and the highest skin dose is from Sr-90. Cerium-144 is also an important contributor.

Figure C-4 shows the total annual doses estimated from episodic and operational releases for each year for the infant EDE only. As can be seen, the episodic releases add no more than 1% to the operational releases except in 1955, 1956, 1958, 1959, 1960, and 1961. The largest EDE from a single release was for IET #3 in 1956. This release accounted for almost 90% of the estimated EDE to the infant during the year. Because meteorological conditions during the IET #3 release were very different from average annual conditions, this release also resulted in a change in the location of maximum exposure during 1956. The operational doses given in Table 4-1 for 1956 are for individuals located in Atomic City. The episodic doses given in Table 4-2 for IET #3 and the combined doses given in Table 4-3 for 1956 are for individuals located at "Building." The procedure for determining the location

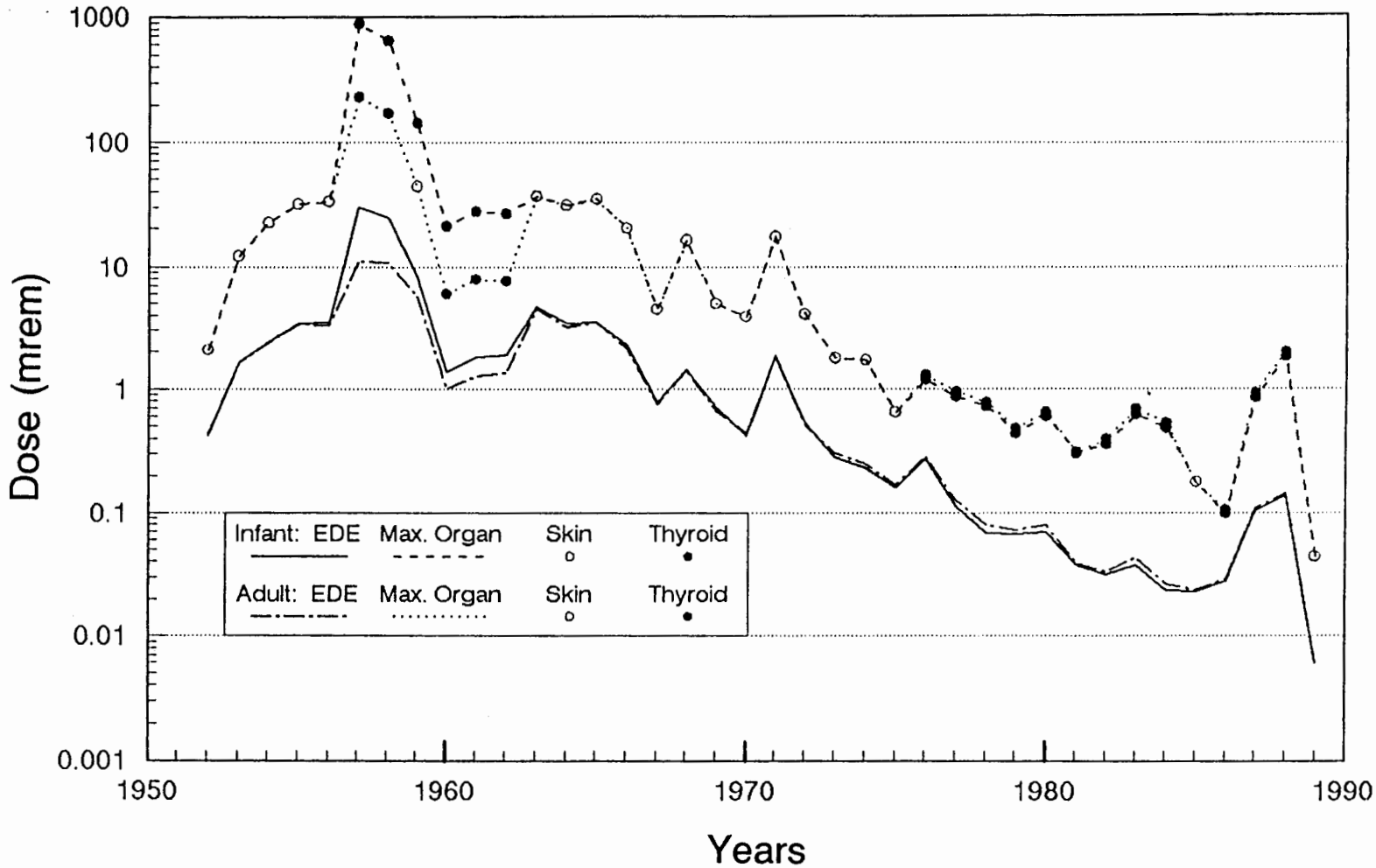

Figure C-3. Comparison of infant and adult operational dose estimates.

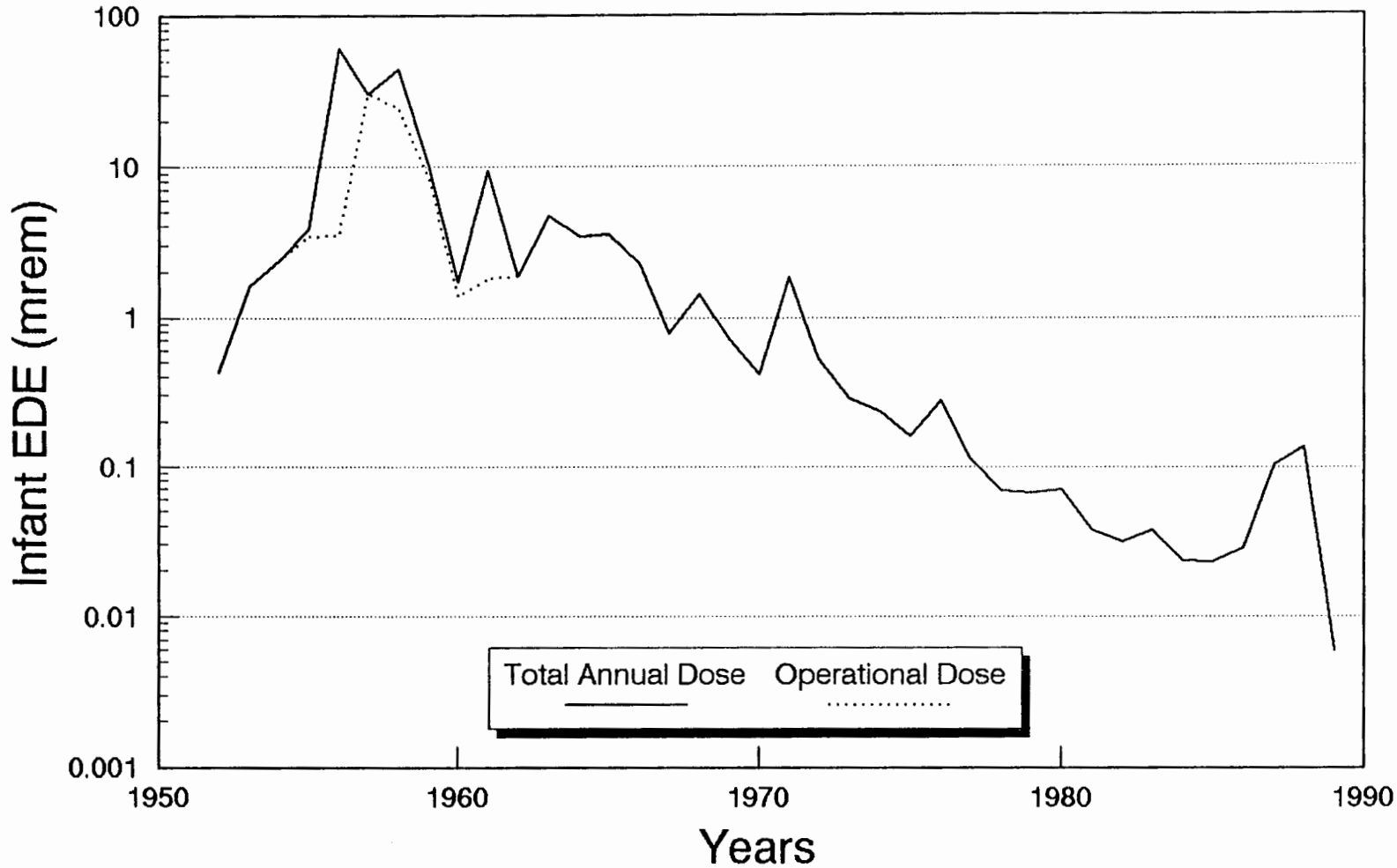

Figure C-4. Annual operational and episodic dose estimates (infant EDE).

of largest total dose is given in Section C.3. After IET #3, the three episodic releases that gave the largest offsite dose were the Fuel Element Burn Test, B in 1957, with an infant EDE of 6 mrem (about 13% of the annual total); FPFRT-8 in 1957, with an infant EDE of 17 mrem (about 38% of the annual total); and the SL-1 Accident in 1961, with an infant EDE of 7 mrem (about 80% of the annual total).

Actual radiation doses from the SL-1 Accident are likely to be lower than those estimated in this evaluation. The largest estimated doses from the SL-1 Accident were from iodine-131 through the ingestion pathway. These estimates were made for Atomic City because it was the location having the highest air concentration from SL-1 Accident releases. However, at the time of the accident, the nearest dairy cow was located at Taber, where air concentrations were much lower.

As shown in Table C-21, the estimated radiation doses were within applicable radiation standards and guidelines in effect at that time for every year of INEL operation. In general, potential offsite doses from INEL activities have been small. The largest radiation doses were calculated for an infant in 1956, when the EDE from operational and episodic releases was estimated to be 61 mrem and the thyroid was estimated to be 1350 mrem. Doses from operational and episodic releases during the late 1950s may have been as high as 9% of the applicable whole body standard in effect at that time and 90% of the standard for doses to any single organ of the body. The trend in recent years has been toward smaller doses as total INEL Site releases have declined. Even with the more restrictive standards in force from 1985 to 1989, offsite radiation doses to a hypothetical individual for those 5 years have been only about 1% of the whole body dose standard and less than 3% of the organ dose standard. Episodic releases have made a substantial contribution to the total potential radiation dose only for a few years during the test programs of the 1950s and early 1960s.

The EDE represents the best available term for comparing the relative importance of different episodic and operational releases. Although the EDE is not exactly equal to the whole body dose equivalent used in the radiation

protection standards in previous years, the EDE provides a reasonable estimate of the whole body dose equivalent.

While this study has tended toward conservative dose estimates, there are uncertainties in the estimates of amounts of radionuclides released and transported offsite, particularly for early years of INEL Site operation. Therefore, these estimates are not sufficiently accurate to ensure that the highest values (i.e., infant thyroid doses for 1956) did not exceed the 1500 mrem/yr guideline.

Tables C-22 through C-38 present examples of some of the detailed dose calculation results for the infant and adult (child for 1988 only). The years of 1957, 1960, 1971, and 1988 were chosen from the operational releases and the IET #3, FPFRT-8, SL-1 Accident, and IET #23 (FEET #2) were chosen from the episodic releases. Operational releases of 1957 are included because it is the year with the highest EDE and organ doses and it includes releases from RaLa. The sample calculation used 1960 operational release data, and IET #23 (FEET #2) is a part of the sample calculation. The operational and episodic releases of 1960 also demonstrate the I-131 contribution to the dose calculation; however, other radionuclides are important. Operational releases of 1971 were chosen because it was one of the years when the skin dose was important. Releases of 1988 show the effect on doses of both I-129 and Sb-125, and it is typical of current operations. IET #3, FPFRT-8, and the SL-1 Accident are the episodic releases with three of the highest estimated EDEs. As mentioned previously, detailed results of the dose calculation for all operational and episodic releases are included in the microfiche at the end of this volume of the report.

The dose calculation results included in the microfiche are arranged as follows: (1) operational releases in reverse chronological order by receptor: adult, child, and infant and (2) episodic releases in reverse chronological order by receptor: adult, child, and infant. The detailed results of each dose calculation are presented on eight sheets that contain the following information:

1. Identification and input parameters

2. Source term, average air concentrations, and some nuclide-dependent input parameters

3. Summary of dose equivalents by radionuclide

4. EDE and six organ dose equivalents due to air immersion

5. EDE and organ dose equivalents due to inhalation

6. EDE and organ dose equivalents due to contamination on the ground surface

7. EDE and organ dose equivalents due to ingestion

8. Summary table of the dose equivalents from items 4 through 7 above.

Table C-22. Dose evaluation for 1988 operational releases (adult)

Report Date: 15-Apr-91
FileName: IDA13A.WK1

Operational Releases

Calendar Year..... 1988
Receptor Location..... Terreton
Receptor Designation..... Adult
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

9.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
54.0 : Distance to Receptor (km)
28.1 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
9.59E-01 : Breathing Rate (m**3/hr) 1.38E-87
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

6.08E+03

Report Date: 15-Apr-91

1988 Operational Releases

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	1.14E-02	3.33E-01	1.26E-02	1.01E-02	9.52E-03	1.11E-02	1.02E-02
Inhalation	2.52E-03	4.34E-04	1.34E-02	1.35E-02	7.24E-04	1.37E-03	5.98E-04
Grnd Surface	6.80E-02	1.41E-01	7.30E-02	5.60E-02	5.16E-02	6.81E-02	5.49E-02
Ingestion	6.52E-02	4.12E-03	1.90E+00	7.30E-03	4.13E-03	3.42E-03	7.39E-03
TOTALS	1.47E-01	4.79E-01	2.00E+00	8.69E-02	6.59E-02	8.40E-02	7.31E-02

Table C-22. (continued).

Report Date: 15-Apr-91
Adult receptor

1988 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	7.0E+02	8.20E-13	7.19E-09	0.00E+00
C-14	5.02E+07	2.7E+00	3.16E-15	2.77E-11	0.00E+00
Na-24	1.50E+01	8.9E-03	9.54E-18	8.36E-14	1.80E-03
Ar-41	1.83E+00	2.1E+03	1.19E-12	1.04E-08	0.00E+00
Cr-51	6.65E+02	1.0E-02	1.17E-17	1.02E-13	1.80E-03
Br-82	3.53E+01	2.2E-02	2.48E-17	2.18E-13	1.80E-03
Kr-85	9.40E+04	1.7E+05	1.99E-10	1.75E-06	0.00E+00
Kr-85m	4.48E+00	3.8E+01	3.31E-14	2.90E-10	0.00E+00
Kr-87	1.27E+00	7.7E+01	3.17E-14	2.77E-10	0.00E+00
Kr-88	2.84E+00	7.4E+01	5.42E-14	4.75E-10	0.00E+00
Rb-89	2.57E-01	8.2E-01	5.43E-18	4.76E-14	1.80E-03
Sr-90	2.51E+05	3.4E-04	3.98E-19	3.49E-15	1.80E-03
Y-91m	8.29E-01	1.8E-03	4.22E-19	3.70E-15	1.80E-03
Tc-99m	6.02E+00	3.7E-03	3.47E-18	3.04E-14	1.80E-03
Ru-106	8.84E+03	1.9E-01	2.22E-16	1.95E-12	1.80E-03
Sb-125	2.43E+04	7.4E+00	8.67E-15	7.60E-11	1.80E-03
I-129 (ELE)	1.38E+11	7.3E-02	8.55E-17	7.49E-13	1.80E-02
I-129 (ORG)	1.38E+11	1.5E-01	1.76E-16	1.54E-12	1.80E-04
I-131	1.93E+02	1.7E-04	1.98E-19	1.73E-15	1.80E-02
I-132	2.30E+00	2.0E-03	1.31E-18	1.15E-14	1.80E-02
I-133	2.08E+01	5.8E-04	6.37E-19	5.59E-15	1.80E-02
Xe-133	1.26E+02	4.5E+02	5.22E-13	4.57E-09	0.00E+00
Xe-135	9.11E+00	1.9E+02	1.92E-13	1.69E-09	0.00E+00
Xe-135m	2.56E-01	4.3E+01	2.77E-16	2.43E-12	0.00E+00
Xe-138	2.36E-01	2.0E+02	8.20E-16	7.19E-12	0.00E+00
Cs-137	2.65E+05	6.8E-04	7.96E-19	6.98E-15	1.80E-03
Cs-138	5.37E-01	8.3E-01	8.13E-17	7.13E-13	1.80E-03
Ba-139	1.39E+00	8.4E-02	3.76E-17	3.30E-13	1.80E-03
U-238	3.91E+13	1.3E-05	1.52E-20	1.33E-16	1.80E-03
Pu-238	7.69E+05	1.4E-05	1.64E-20	1.44E-16	1.80E-03
Pu-239/240	2.12E+08	2.1E-06	2.46E-21	2.16E-17	1.80E-03
TOTAL		1.74E+05			

Table C-22. (continued).

Report Date: 15-Apr-91
Adult receptor

1988 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	4.55E-03	4.55E-03	4.55E-03	4.55E-03	4.55E-03	0.00E+00	4.55E-03
C-14	3.23E-03	9.20E-08	3.23E-03	3.23E-03	0.00E+00	3.23E-03	3.23E-03
Na-24	6.45E-07	1.27E-06	5.95E-07	8.57E-07	4.75E-07	5.31E-07	5.79E-07
Ar-41	7.86E-03	1.45E-02	8.70E-03	6.98E-03	6.60E-03	7.46E-03	7.04E-03
Cr-51	2.64E-07	2.33E-07	1.89E-07	3.45E-07	1.38E-07	1.90E-07	1.75E-07
Br-82	2.22E-06	2.51E-06	2.02E-06	3.02E-06	1.51E-06	2.18E-06	1.91E-06
Kr-85	2.23E-03	3.14E-01	2.41E-03	1.94E-03	1.81E-03	2.27E-03	1.98E-03
Kr-85m	2.70E-05	8.66E-05	2.98E-05	2.22E-05	2.04E-05	3.27E-05	2.13E-05
Kr-87	1.41E-04	5.22E-04	1.51E-04	1.25E-04	1.19E-04	1.36E-04	1.28E-04
Kr-88	8.30E-04	2.27E-03	8.91E-04	7.42E-04	7.05E-04	7.84E-04	7.55E-04
Rb-89	6.53E-08	1.37E-07	6.92E-08	6.77E-08	5.34E-08	6.02E-08	5.74E-08
Sr-90	2.06E-05	4.92E-04	0.00E+00	3.68E-05	0.00E+00	2.00E-04	8.75E-05
Y-91m	1.37E-09	1.84E-09	1.37E-09	2.02E-09	1.02E-09	1.27E-09	1.12E-09
Tc-99m	5.26E-09	5.40E-09	1.01E-08	5.98E-09	3.17E-09	5.48E-09	3.58E-09
Ru-106	1.24E-03	2.01E-02	2.95E-04	7.16E-03	2.21E-04	2.72E-04	2.42E-04
Sb-125	6.53E-02	1.11E-01	6.89E-02	5.98E-02	4.98E-02	6.45E-02	5.39E-02
I-129 (ELE)	5.97E-02	9.06E-03	1.86E+00	1.84E-03	1.45E-03	3.21E-03	6.90E-04
I-129 (ORG)	1.49E-03	1.86E-04	4.69E-02	3.77E-05	2.97E-05	6.59E-05	1.42E-05
I-131	1.57E-06	2.65E-07	4.99E-05	8.69E-08	8.07E-08	1.07E-07	8.95E-08
I-132	6.45E-08	1.75E-07	1.37E-07	6.45E-08	5.02E-08	5.94E-08	5.44E-08
I-133	1.34E-07	3.83E-07	2.71E-06	4.83E-08	4.51E-08	5.55E-08	4.93E-08
Xe-133	9.17E-05	3.03E-04	1.05E-04	6.77E-05	6.05E-05	1.20E-04	4.42E-05
Xe-135	2.40E-04	6.99E-04	2.61E-04	2.02E-04	1.87E-04	2.69E-04	2.06E-04
Xe-135m	5.96E-07	9.89E-07	6.46E-07	5.18E-07	4.82E-07	6.04E-07	5.29E-07
Xe-138	5.14E-06	1.06E-05	5.53E-06	4.56E-06	4.32E-06	4.96E-06	4.62E-06
Cs-137	6.47E-05	1.74E-04	6.97E-05	5.76E-05	4.47E-05	6.38E-05	5.85E-05
Cs-138	1.15E-06	2.62E-06	1.20E-06	1.38E-06	9.21E-07	1.04E-06	9.89E-07
Ba-139	5.87E-08	5.96E-07	8.94E-09	3.01E-07	6.14E-09	9.57E-09	6.41E-09
U-238	1.61E-05	2.74E-05	7.43E-08	1.28E-04	5.00E-08	1.03E-05	7.40E-07
Pu-238	6.47E-05	1.95E-08	4.10E-10	3.31E-10	2.53E-04	1.14E-03	9.42E-05
Pu-239/240	1.08E-05	1.38E-09	5.72E-11	4.41E-11	4.22E-05	1.96E-04	1.56E-05

Table C-22. (continued).

Report Date: 15-Apr-91
Adult receptor

1988 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	9.20E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.28E-07	3.34E-07	2.42E-07	2.03E-07	1.93E-07	2.12E-07	2.07E-07
Ar-41	7.86E-03	1.45E-02	8.70E-03	6.98E-03	6.60E-03	7.46E-03	7.04E-03
Cr-51	1.85E-09	2.39E-09	2.00E-09	1.58E-09	1.46E-09	2.01E-09	1.62E-09
Br-82	3.40E-07	4.71E-07	3.74E-07	3.00E-07	2.80E-07	3.32E-07	3.03E-07
Kr-85	2.23E-03	3.14E-01	2.41E-03	1.94E-03	1.81E-03	2.27E-03	1.98E-03
Kr-85m	2.70E-05	8.66E-05	2.98E-05	2.22E-05	2.04E-05	3.27E-05	2.13E-05
Kr-87	1.41E-04	5.22E-04	1.51E-04	1.25E-04	1.19E-04	1.36E-04	1.28E-04
Kr-88+D	8.30E-04	2.27E-03	8.91E-04	7.42E-04	7.05E-04	7.84E-04	7.55E-04
Rb-89	6.18E-08	1.25E-07	6.73E-08	5.48E-08	5.20E-08	5.86E-08	5.59E-08
Sr-90+D	0.00E+00	3.38E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.13E-09	1.60E-09	1.23E-09	9.88E-10	9.20E-10	1.14E-09	1.01E-09
Tc-99m	2.30E-09	2.92E-09	2.57E-09	1.87E-09	1.69E-09	2.93E-09	1.70E-09
Ru-106+D	2.33E-07	2.85E-06	2.56E-07	2.05E-07	1.91E-07	2.33E-07	2.09E-07
Sb-125+D	1.89E-05	2.97E-05	2.05E-05	1.62E-05	1.50E-05	1.92E-05	1.64E-05
I-129 (ELE)	3.56E-09	9.91E-09	3.55E-09	1.76E-09	1.41E-09	3.09E-09	6.68E-10
I-129 (ORG)	7.32E-09	2.04E-08	7.29E-09	3.62E-09	2.90E-09	6.36E-09	1.37E-09
I-131	3.77E-10	6.96E-10	4.09E-10	3.24E-10	3.00E-10	3.99E-10	3.34E-10
I-132	1.56E-08	2.56E-08	1.72E-08	1.38E-08	1.29E-08	1.52E-08	1.39E-08
I-133	1.95E-09	4.44E-09	2.13E-09	1.71E-09	1.60E-09	1.96E-09	1.74E-09
Xe-133	9.17E-05	3.03E-04	1.05E-04	6.77E-05	6.05E-05	1.20E-04	4.42E-05
Xe-135	2.40E-04	6.99E-04	2.61E-04	2.02E-04	1.87E-04	2.69E-04	2.06E-04
Xe-135m	5.96E-07	9.89E-07	6.46E-07	5.18E-07	4.82E-07	6.04E-07	5.29E-07
Xe-138	5.14E-06	1.06E-05	5.53E-06	4.56E-06	4.32E-06	4.96E-06	4.62E-06
Cs-137+D	2.30E-09	4.16E-09	2.54E-09	2.02E-09	1.89E-09	2.29E-09	2.06E-09
Cs-138	1.03E-06	2.16E-06	1.13E-06	9.18E-07	8.69E-07	9.83E-07	9.34E-07
Ba-139	6.77E-09	2.71E-07	7.48E-09	5.60E-09	5.15E-09	8.01E-09	5.37E-09
U-238+D	1.51E-12	1.51E-12	1.00E-10	1.70E-12	1.26E-12	1.17E-12	1.70E-12
Pu-238	7.23E-15	5.01E-14	2.80E-15	1.93E-15	1.17E-15	3.42E-15	8.70E-16
Pu-239/240	1.01E-15	3.56E-15	8.33E-16	5.95E-16	5.04E-16	9.86E-16	4.79E-16
TOTALS	1.14E-02	3.33E-01	1.26E-02	1.01E-02	9.52E-03	1.11E-02	1.02E-02

Table C-22. (continued).

Report Date: 15-Apr-91
Adult receptor

1988 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	4.34E-04	4.34E-04	4.34E-04	4.34E-04	4.34E-04	0.00E+00	4.34E-04
C-14	5.58E-05	0.00E+00	5.58E-05	5.58E-05	0.00E+00	5.58E-05	5.58E-05
Na-24	7.62E-08	0.00E+00	0.00E+00	3.53E-07	0.00E+00	0.00E+00	6.25E-08
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	2.55E-08	0.00E+00	0.00E+00	1.97E-07	0.00E+00	0.00E+00	0.00E+00
Br-82	2.71E-07	0.00E+00	0.00E+00	1.31E-06	0.00E+00	2.92E-07	1.69E-07
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.69E-09	0.00E+00	0.00E+00	1.14E-08	0.00E+00	0.00E+00	0.00E+00
Sr-90	4.35E-06	0.00E+00	0.00E+00	3.68E-05	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.10E-10	0.00E+00	0.00E+00	9.23E-10	0.00E+00	0.00E+00	0.00E+00
Tc-99m	9.34E-10	0.00E+00	5.26E-09	2.48E-09	0.00E+00	0.00E+00	3.80E-10
Ru-106	8.23E-04	0.00E+00	0.00E+00	6.92E-03	0.00E+00	0.00E+00	0.00E+00
Sb-125	7.14E-04	0.00E+00	0.00E+00	5.90E-03	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.29E-04	0.00E+00	4.24E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	2.66E-04	0.00E+00	8.71E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	5.32E-08	0.00E+00	1.83E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	3.64E-09	0.00E+00	6.95E-08	1.10E-08	0.00E+00	0.00E+00	0.00E+00
I-133	2.89E-08	0.00E+00	9.64E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	2.14E-07	0.00E+00	1.94E-07	2.21E-07	0.00E+00	1.94E-07	2.08E-07
Cs-138	6.01E-08	0.00E+00	0.00E+00	4.03E-07	0.00E+00	0.00E+00	0.00E+00
Ba-139	5.06E-08	0.00E+00	0.00E+00	2.94E-07	0.00E+00	0.00E+00	0.00E+00
U-238	1.54E-05	0.00E+00	0.00E+00	1.28E-04	0.00E+00	0.00E+00	0.00E+00
Pu-238	6.34E-05	0.00E+00	0.00E+00	0.00E+00	2.48E-04	1.12E-03	9.23E-05
Pu-239/240	1.05E-05	0.00E+00	0.00E+00	0.00E+00	4.13E-05	1.92E-04	1.53E-05
TOTALS	2.52E-03	4.34E-04	1.34E-02	1.35E-02	7.24E-04	1.37E-03	5.98E-04

Table C-22. (continued).

Report Date: 15-Apr-91
Adult receptor

1988 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	3.32E-07	9.34E-07	3.53E-07	2.95E-07	2.82E-07	3.09E-07	3.02E-07
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.73E-07	2.30E-07	1.87E-07	1.47E-07	1.36E-07	1.88E-07	1.52E-07
Br-82	1.49E-06	2.04E-06	1.64E-06	1.31E-06	1.23E-06	1.46E-06	1.33E-06
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.74E-09	1.21E-08	1.90E-09	1.54E-09	1.46E-09	1.65E-09	1.57E-09
Sr-90+D	0.00E+00	4.92E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.25E-10	2.36E-10	1.36E-10	1.09E-10	1.02E-10	1.26E-10	1.11E-10
Tc-99m	2.02E-09	2.49E-09	2.24E-09	1.63E-09	1.48E-09	2.56E-09	1.49E-09
Ru-106+D	2.70E-04	2.01E-02	2.95E-04	2.37E-04	2.21E-04	2.71E-04	2.42E-04
Sb-125+D	6.39E-02	1.11E-01	6.89E-02	5.39E-02	4.98E-02	6.45E-02	5.39E-02
I-129 (ELE)	3.77E-03	9.06E-03	3.72E-03	1.84E-03	1.45E-03	3.21E-03	6.90E-04
I-129 (ORG)	7.76E-05	1.86E-04	7.65E-05	3.77E-05	2.97E-05	6.59E-05	1.42E-05
I-131	1.01E-07	2.65E-07	1.09E-07	8.66E-08	8.04E-08	1.07E-07	8.91E-08
I-132	4.52E-08	1.50E-07	4.99E-08	3.97E-08	3.73E-08	4.42E-08	4.05E-08
I-133	5.32E-08	3.78E-07	5.81E-08	4.66E-08	4.35E-08	5.36E-08	4.76E-08
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	5.47E-05	1.74E-04	6.01E-05	4.80E-05	4.47E-05	5.42E-05	4.89E-05
Cs-138	6.14E-08	4.57E-07	6.68E-08	5.45E-08	5.17E-08	5.85E-08	5.51E-08
Ba-139	1.32E-09	3.26E-07	1.46E-09	1.09E-09	9.95E-10	1.57E-09	1.04E-09
U-238+D	6.89E-08	2.74E-05	7.42E-08	5.46E-08	5.00E-08	7.63E-08	4.78E-08
Pu-238	2.33E-09	1.95E-08	4.10E-10	3.31E-10	7.14E-11	5.56E-10	1.22E-10
Pu-239/240	1.86E-10	1.38E-09	5.72E-11	4.41E-11	2.39E-11	7.25E-11	2.75E-11
TOTALS	6.80E-02	1.41E-01	7.30E-02	5.60E-02	5.16E-02	6.81E-02	5.49E-02

Table C-22. (continued).

Report Date: 15-Apr-91
Adult receptor

1988 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	4.12E-03	4.12E-03	4.12E-03	4.12E-03	4.12E-03	0.00E+00	4.12E-03
C-14	3.17E-03	0.00E+00	3.17E-03	3.17E-03	0.00E+00	3.17E-03	3.17E-03
Na-24	8.42E-09	0.00E+00	0.00E+00	5.77E-09	0.00E+00	1.02E-08	8.42E-09
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	6.28E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.13E-08
Br-82	1.19E-07	0.00E+00	0.00E+00	9.79E-08	0.00E+00	9.79E-08	1.05E-07
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	5.71E-40	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.62E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.00E-04	8.75E-05
Y-91m	1.56E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	5.12E-12	0.00E+00	2.65E-11	0.00E+00	0.00E+00	0.00E+00	1.96E-12
Ru-106	1.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-125	7.46E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	5.58E-02	0.00E+00	1.85E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	1.15E-03	0.00E+00	3.81E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	1.41E-06	0.00E+00	4.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	7.42E-13	0.00E+00	1.82E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	4.96E-08	0.00E+00	1.69E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	9.81E-06	0.00E+00	9.42E-06	9.42E-06	0.00E+00	9.42E-06	9.42E-06
Cs-138	1.45E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	7.38E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-238	6.37E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.02E-05	6.92E-07
Pu-238	1.27E-06	0.00E+00	0.00E+00	0.00E+00	5.01E-06	2.24E-05	1.87E-06
Pu-239/240	2.15E-07	0.00E+00	0.00E+00	0.00E+00	8.02E-07	3.91E-06	2.96E-07
TOTALS	6.52E-02	4.12E-03	1.90E+00	7.30E-03	4.13E-03	3.42E-03	7.39E-03

Table C-23. Dose evaluation for 1988 operational releases (child)

Report Date: 15-Apr-91
File Name: IDA13C.WK1

Operational Releases

Calendar Year..... 1988
Receptor Location..... Terreton
Receptor Designation..... Child
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

9.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
54.0 : Distance to Receptor (km)
28.1 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
6.25E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 15-Apr-91
File Name: IDA13C.WK1

1988 Operational Releases

Summary of Dose Equivalents to Child From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	1.14E-02	3.33E-01	1.26E-02	1.01E-02	9.52E-03	1.11E-02	1.02E-02
Inhalation	2.48E-03	3.78E-04	1.31E-02	1.42E-02	5.73E-04	1.09E-03	4.27E-04
Grnd Surface	6.80E-02	1.41E-01	7.30E-02	5.60E-02	5.16E-02	6.81E-02	5.49E-02
Ingestion	8.27E-02	8.30E-03	2.57E+00	8.35E-03	8.29E-03	8.75E-03	8.41E-03
TOTALS	1.65E-01	4.83E-01	2.66E+00	8.87E-02	6.99E-02	8.90E-02	7.39E-02

Table C-23. (continued).

Report Date: 15-Apr-91
Child receptor

1988 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	7.0E+02	8.20E-13	7.19E-09	0.00E+00
C-14	5.02E+07	2.7E+00	3.16E-15	2.77E-11	0.00E+00
Na-24	1.50E+01	8.9E-03	9.54E-18	8.36E-14	1.80E-03
Ar-41	1.83E+00	2.1E+03	1.19E-12	1.04E-08	0.00E+00
Cr-51	6.65E+02	1.0E-02	1.17E-17	1.02E-13	1.80E-03
Br-82	3.53E+01	2.2E-02	2.48E-17	2.18E-13	1.80E-03
Kr-85	9.40E+04	1.7E+05	1.99E-10	1.75E-06	0.00E+00
Kr-85m	4.48E+00	3.8E+01	3.31E-14	2.90E-10	0.00E+00
Kr-87	1.27E+00	7.7E+01	3.17E-14	2.77E-10	0.00E+00
Kr-88	2.84E+00	7.4E+01	5.42E-14	4.75E-10	0.00E+00
Rb-89	2.57E-01	8.2E-01	5.43E-18	4.76E-14	1.80E-03
Sr-90	2.51E+05	3.4E-04	3.98E-19	3.49E-15	1.80E-03
Y-91m	8.29E-01	1.8E-03	4.22E-19	3.70E-15	1.80E-03
Tc-99m	6.02E+00	3.7E-03	3.47E-18	3.04E-14	1.80E-03
Ru-106	8.84E+03	1.9E-01	2.22E-16	1.95E-12	1.80E-03
Sb-125	2.43E+04	7.4E+00	8.67E-15	7.60E-11	1.80E-03
I-129 (ELE)	1.38E+11	7.3E-02	8.55E-17	7.49E-13	1.80E-02
I-129 (ORG)	1.38E+11	1.5E-01	1.76E-16	1.54E-12	1.80E-04
I-131	1.93E+02	1.7E-04	1.98E-19	1.73E-15	1.80E-02
I-132	2.30E+00	2.0E-03	1.31E-18	1.15E-14	1.80E-02
I-133	2.08E+01	5.8E-04	6.37E-19	5.59E-15	1.80E-02
Xe-133	1.26E+02	4.5E+02	5.22E-13	4.57E-09	0.00E+00
Xe-135	9.11E+00	1.9E+02	1.92E-13	1.69E-09	0.00E+00
Xe-135m	2.56E-01	4.3E+01	2.77E-16	2.43E-12	0.00E+00
Xe-138	2.36E-01	2.0E+02	8.20E-16	7.19E-12	0.00E+00
Cs-137	2.65E+05	6.8E-04	7.96E-19	6.98E-15	1.80E-03
Cs-138	5.37E-01	8.3E-01	8.13E-17	7.13E-13	1.80E-03
Ba-139	1.39E+00	8.4E-02	3.76E-17	3.30E-13	1.80E-03
U-238	3.91E+13	1.3E-05	1.52E-20	1.33E-16	1.80E-03
Pu-238	7.69E+05	1.4E-05	1.64E-20	1.44E-16	1.80E-03
Pu-239/240	2.12E+08	2.1E-06	2.46E-21	2.16E-17	1.80E-03
TOTAL		1.74E+05			

Table C-23. (continued).

Report Date: 15-Apr-91
Child receptor

1988 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	4.58E-03	4.58E-03	4.58E-03	4.58E-03	4.58E-03	4.58E-03	4.58E-03
C-14	3.94E-03	3.94E-03	3.94E-03	3.94E-03	3.94E-03	3.94E-03	3.94E-03
Na-24	6.16E-07	1.27E-06	5.95E-07	7.32E-07	4.75E-07	5.28E-07	5.56E-07
Ar-41	7.86E-03	1.45E-02	8.70E-03	6.98E-03	6.60E-03	7.46E-03	7.04E-03
Cr-51	2.41E-07	2.33E-07	1.89E-07	2.77E-07	1.38E-07	1.90E-07	1.70E-07
Br-82	2.10E-06	2.51E-06	2.02E-06	2.54E-06	1.51E-06	2.05E-06	1.83E-06
Kr-85	2.23E-03	3.14E-01	2.41E-03	1.94E-03	1.81E-03	2.27E-03	1.98E-03
Kr-85m	2.70E-05	8.66E-05	2.98E-05	2.22E-05	2.04E-05	3.27E-05	2.13E-05
Kr-87	1.41E-04	5.22E-04	1.51E-04	1.25E-04	1.19E-04	1.36E-04	1.28E-04
Kr-88	8.30E-04	2.27E-03	8.91E-04	7.42E-04	7.05E-04	7.84E-04	7.55E-04
Rb-89	6.47E-08	1.37E-07	6.92E-08	6.38E-08	5.34E-08	6.02E-08	5.74E-08
Sr-90	1.89E-05	4.94E-04	1.10E-06	1.12E-06	1.10E-06	2.43E-04	7.93E-05
Y-91m	1.33E-09	1.84E-09	1.37E-09	1.70E-09	1.02E-09	1.27E-09	1.12E-09
Tc-99m	4.93E-09	5.40E-09	8.25E-09	5.11E-09	3.17E-09	5.48E-09	3.44E-09
Ru-106	1.72E-03	2.01E-02	3.50E-04	1.02E-02	2.82E-04	3.34E-04	3.10E-04
Sb-125	6.48E-02	1.11E-01	6.89E-02	5.77E-02	4.98E-02	6.45E-02	5.39E-02
I-129 (ELE)	7.62E-02	9.16E-03	2.51E+00	1.99E-03	1.53E-03	3.50E-03	8.08E-04
I-129 (ORG)	1.82E-03	1.89E-04	6.01E-02	4.30E-05	3.18E-05	7.29E-05	1.70E-05
I-131	2.45E-06	2.74E-07	8.08E-05	1.10E-07	8.80E-08	1.23E-07	1.01E-07
I-132	6.67E-08	1.76E-07	1.69E-07	6.89E-08	5.11E-08	6.02E-08	5.51E-08
I-133	1.05E-07	3.83E-07	1.73E-06	4.83E-08	4.51E-08	5.55E-08	4.93E-08
Xe-133	9.17E-05	3.03E-04	1.05E-04	6.77E-05	6.05E-05	1.20E-04	4.42E-05
Xe-135	2.40E-04	6.99E-04	2.61E-04	2.02E-04	1.87E-04	2.69E-04	2.06E-04
Xe-135m	5.96E-07	9.89E-07	6.46E-07	5.18E-07	4.82E-07	6.04E-07	5.29E-07
Xe-138	5.14E-06	1.06E-05	5.53E-06	4.56E-06	4.32E-06	4.96E-06	4.62E-06
Cs-137	6.03E-05	1.78E-04	6.57E-05	5.33E-05	5.03E-05	6.05E-05	5.52E-05
Cs-138	1.13E-06	2.62E-06	1.20E-06	1.24E-06	9.21E-07	1.04E-06	9.89E-07
Ba-139	4.11E-08	5.96E-07	8.94E-09	1.98E-07	6.14E-09	9.57E-09	6.41E-09
U-238	1.07E-05	2.74E-05	7.43E-08	8.35E-05	5.00E-08	9.68E-06	6.96E-07
Pu-238	3.71E-05	7.61E-07	7.42E-07	1.39E-05	1.62E-04	6.07E-04	3.71E-05
Pu-239/240	6.07E-06	1.18E-07	1.16E-07	2.00E-06	2.79E-05	1.06E-04	6.07E-06
TOTALS	1.65E-01	4.83E-01	2.66E+00	8.87E-02	6.99E-02	8.90E-02	7.39E-02

Table C-23. (continued).

Report Date: 15-Apr-91
Child receptor

1988 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	9.20E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.28E-07	3.34E-07	2.42E-07	2.03E-07	1.93E-07	2.12E-07	2.07E-07
Ar-41	7.86E-03	1.45E-02	8.70E-03	6.98E-03	6.60E-03	7.46E-03	7.04E-03
Cr-51	1.85E-09	2.39E-09	2.00E-09	1.58E-09	1.46E-09	2.01E-09	1.62E-09
Br-82	3.40E-07	4.71E-07	3.74E-07	3.00E-07	2.80E-07	3.32E-07	3.03E-07
Kr-85	2.23E-03	3.14E-01	2.41E-03	1.94E-03	1.81E-03	2.27E-03	1.98E-03
Kr-85m	2.70E-05	8.66E-05	2.98E-05	2.22E-05	2.04E-05	3.27E-05	2.13E-05
Kr-87	1.41E-04	5.22E-04	1.51E-04	1.25E-04	1.19E-04	1.36E-04	1.28E-04
Kr-88+D	8.30E-04	2.27E-03	8.91E-04	7.42E-04	7.05E-04	7.84E-04	7.55E-04
Rb-89	6.18E-08	1.25E-07	6.73E-08	5.48E-08	5.20E-08	5.86E-08	5.59E-08
Sr-90+D	0.00E+00	3.38E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.13E-09	1.60E-09	1.23E-09	9.88E-10	9.20E-10	1.14E-09	1.01E-09
Tc-99m	2.30E-09	2.92E-09	2.57E-09	1.87E-09	1.69E-09	2.93E-09	1.70E-09
Ru-106+D	2.33E-07	2.85E-06	2.56E-07	2.05E-07	1.91E-07	2.33E-07	2.09E-07
Sb-125+D	1.89E-05	2.97E-05	2.05E-05	1.62E-05	1.50E-05	1.92E-05	1.64E-05
I-129 (ELE)	3.56E-09	9.91E-09	3.55E-09	1.76E-09	1.41E-09	3.09E-09	6.68E-10
I-129 (ORG)	7.32E-09	2.04E-08	7.29E-09	3.62E-09	2.90E-09	6.36E-09	1.37E-09
I-131	3.77E-10	6.96E-10	4.09E-10	3.24E-10	3.00E-10	3.99E-10	3.34E-10
I-132	1.56E-08	2.56E-08	1.72E-08	1.38E-08	1.29E-08	1.52E-08	1.39E-08
I-133	1.95E-09	4.44E-09	2.13E-09	1.71E-09	1.60E-09	1.96E-09	1.74E-09
Xe-133	9.17E-05	3.03E-04	1.05E-04	6.77E-05	6.05E-05	1.20E-04	4.42E-05
Xe-135	2.40E-04	6.99E-04	2.61E-04	2.02E-04	1.87E-04	2.69E-04	2.06E-04
Xe-135m	5.96E-07	9.89E-07	6.46E-07	5.18E-07	4.82E-07	6.04E-07	5.29E-07
Xe-138	5.14E-06	1.06E-05	5.53E-06	4.56E-06	4.32E-06	4.96E-06	4.62E-06
Cs-137+D	2.30E-09	4.16E-09	2.54E-09	2.02E-09	1.89E-09	2.29E-09	2.06E-09
Cs-138	1.03E-06	2.16E-06	1.13E-06	9.18E-07	8.69E-07	9.83E-07	9.34E-07
Ba-139	6.77E-09	2.71E-07	7.48E-09	5.60E-09	5.15E-09	8.01E-09	5.37E-09
U-238+D	1.51E-12	1.51E-12	1.00E-10	1.70E-12	1.26E-12	1.17E-12	1.70E-12
Pu-238	7.23E-15	5.01E-14	2.80E-15	1.93E-15	1.17E-15	3.42E-15	8.70E-16
Pu-239/240	1.01E-15	3.56E-15	8.33E-16	5.95E-16	5.04E-16	9.86E-16	4.79E-16
TOTALS	1.14E-02	3.33E-01	1.26E-02	1.01E-02	9.52E-03	1.11E-02	1.02E-02

Table C-23. (continued).

Report Date: 15-Apr-91
Child receptor

1988 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	3.16E-04	3.16E-04	3.16E-04	3.16E-04	3.16E-04	3.16E-04	3.16E-04
C-14	4.94E-05	4.94E-05	4.94E-05	4.94E-05	4.94E-05	4.94E-05	4.94E-05
Na-24	4.96E-08	0.00E+00	0.00E+00	2.30E-07	0.00E+00	0.00E+00	4.08E-08
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.67E-08	0.00E+00	0.00E+00	1.28E-07	0.00E+00	0.00E+00	0.00E+00
Br-82	1.77E-07	0.00E+00	0.00E+00	8.56E-07	0.00E+00	1.90E-07	1.10E-07
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.10E-09	0.00E+00	0.00E+00	7.43E-09	0.00E+00	0.00E+00	0.00E+00
Sr-90	4.93E-07	2.99E-08	2.99E-08	5.00E-08	2.99E-08	6.70E-06	2.18E-06
Y-91m	7.18E-11	0.00E+00	0.00E+00	6.02E-10	0.00E+00	0.00E+00	0.00E+00
Tc-99m	6.09E-10	0.00E+00	3.43E-09	1.62E-09	0.00E+00	0.00E+00	2.47E-10
Ru-106	1.22E-03	1.17E-05	1.49E-05	9.92E-03	1.94E-05	1.58E-05	1.67E-05
Sb-125	4.65E-04	0.00E+00	0.00E+00	3.85E-03	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.25E-04	1.66E-07	4.16E-03	1.06E-06	1.48E-07	4.87E-07	1.91E-07
I-129 (ORG)	2.56E-04	3.42E-07	8.55E-03	2.18E-06	3.03E-07	1.00E-06	3.92E-07
I-131	8.02E-08	3.09E-10	2.61E-06	5.61E-09	3.04E-10	5.61E-10	4.41E-10
I-132	5.85E-09	4.79E-10	1.01E-07	1.55E-08	9.35E-10	7.19E-10	7.19E-10
I-133	1.89E-08	0.00E+00	6.28E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	1.05E-07	7.90E-08	9.86E-08	1.23E-07	9.86E-08	1.06E-07	1.08E-07
Cs-138	3.92E-08	0.00E+00	0.00E+00	2.63E-07	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.30E-08	0.00E+00	0.00E+00	1.92E-07	0.00E+00	0.00E+00	0.00E+00
U-238	1.00E-05	0.00E+00	0.00E+00	8.34E-05	0.00E+00	0.00E+00	0.00E+00
Pu-238	3.66E-05	7.31E-07	7.31E-07	1.39E-05	1.60E-04	5.98E-04	3.66E-05
Pu-239/240	5.98E-06	1.15E-07	1.15E-07	1.99E-06	2.75E-05	1.05E-04	5.98E-06
TOTAL	2.48E-03	3.78E-04	1.31E-02	1.42E-02	5.73E-04	1.09E-03	4.27E-04

Table C-23. (continued).

Report Date: 15-Apr-91
Child receptor

1988 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	3.32E-07	9.34E-07	3.53E-07	2.95E-07	2.82E-07	3.09E-07	3.02E-07
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.73E-07	2.30E-07	1.87E-07	1.47E-07	1.36E-07	1.88E-07	1.52E-07
Br-82	1.49E-06	2.04E-06	1.64E-06	1.31E-06	1.23E-06	1.46E-06	1.33E-06
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.74E-09	1.21E-08	1.90E-09	1.54E-09	1.46E-09	1.65E-09	1.57E-09
Sr-90	0.00E+00	4.92E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.25E-10	2.36E-10	1.36E-10	1.09E-10	1.02E-10	1.26E-10	1.11E-10
Tc-99m	2.02E-09	2.49E-09	2.24E-09	1.63E-09	1.48E-09	2.56E-09	1.49E-09
Ru-106	2.70E-04	2.01E-02	2.95E-04	2.37E-04	2.21E-04	2.71E-04	2.42E-04
Sb-125	6.39E-02	1.11E-01	6.89E-02	5.39E-02	4.98E-02	6.45E-02	5.39E-02
I-129 (ELE)	3.77E-03	9.06E-03	3.72E-03	1.84E-03	1.45E-03	3.21E-03	6.90E-04
I-129 (ORG)	7.76E-05	1.86E-04	7.65E-05	3.77E-05	2.97E-05	6.59E-05	1.42E-05
I-131	1.01E-07	2.65E-07	1.09E-07	8.66E-08	8.04E-08	1.07E-07	8.91E-08
I-132	4.52E-08	1.50E-07	4.99E-08	3.97E-08	3.73E-08	4.42E-08	4.05E-08
I-133	5.32E-08	3.78E-07	5.81E-08	4.66E-08	4.35E-08	5.36E-08	4.76E-08
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	5.47E-05	1.74E-04	6.01E-05	4.80E-05	4.47E-05	5.42E-05	4.89E-05
Cs-138	6.14E-08	4.57E-07	6.68E-08	5.45E-08	5.17E-08	5.85E-08	5.51E-08
Ba-139	1.32E-09	3.26E-07	1.46E-09	1.09E-09	9.95E-10	1.57E-09	1.04E-09
U-238+D	6.89E-08	2.74E-05	7.42E-08	5.46E-08	5.00E-08	7.63E-08	4.78E-08
Pu-238	2.33E-09	1.95E-08	4.10E-10	3.31E-10	7.14E-11	5.56E-10	1.22E-10
Pu-239/240	1.86E-10	1.38E-09	5.72E-11	4.41E-11	2.39E-11	7.25E-11	2.75E-11
TOTALS	6.80E-02	1.41E-01	7.30E-02	5.60E-02	5.16E-02	6.81E-02	5.49E-02

Table C-23. (continued).

Report Date: 15-Apr-91
Child receptor

1988 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	4.26E-03	4.26E-03	4.26E-03	4.26E-03	4.26E-03	4.26E-03	4.26E-03
C-14	3.89E-03	3.89E-03	3.89E-03	3.89E-03	3.89E-03	3.89E-03	3.89E-03
Na-24	6.47E-09	0.00E+00	0.00E+00	4.43E-09	0.00E+00	7.85E-09	6.47E-09
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	4.95E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.67E-08
Br-82	9.25E-08	0.00E+00	0.00E+00	7.61E-08	0.00E+00	7.61E-08	8.16E-08
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.32E-40	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90+D	1.84E-05	1.07E-06	1.07E-06	1.07E-06	1.07E-06	2.36E-04	7.72E-05
Y-91m	6.34E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	2.30E-12	0.00E+00	1.19E-11	0.00E+00	0.00E+00	0.00E+00	8.80E-13
Ru-106	2.32E-04	4.08E-05	4.08E-05	4.08E-05	4.08E-05	4.63E-05	5.10E-05
Sb-125	4.40E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	7.23E-02	9.86E-05	2.51E+00	1.51E-04	8.55E-05	2.90E-04	1.18E-04
I-129 (ORG)	1.49E-03	2.03E-06	5.15E-02	3.11E-06	1.76E-06	5.95E-06	2.43E-06
I-131	2.26E-06	8.52E-09	7.81E-05	1.70E-08	6.96E-09	1.49E-08	1.14E-08
I-132	7.46E-13	6.46E-14	1.67E-11	1.31E-13	1.10E-13	9.21E-14	9.42E-14
I-133	3.07E-08	0.00E+00	1.04E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	5.53E-06	4.40E-06	5.46E-06	5.18E-06	5.46E-06	6.20E-06	6.20E-06
Cs-138	5.89E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.00E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-238	5.97E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.60E-06	6.49E-07
Pu-238	4.92E-07	1.06E-08	1.06E-08	1.06E-08	2.23E-06	8.37E-06	5.19E-07
Pu-239/240	8.38E-08	1.59E-09	1.59E-09	1.68E-09	3.78E-07	1.42E-06	8.38E-08
TOTALS	8.27E-02	8.30E-03	2.57E+00	8.35E-03	8.29E-03	8.75E-03	8.41E-03

Table C-24. Dose evaluation for 1988 operational releases (infant)

Report Date: 15-Apr-91
FileName: IDA13B.WK1

Operational Releases

Calendar Year..... 1988
Receptor Location..... Terreton
Receptor Designation..... Infant
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

9.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
54.0 : Distance to Receptor (km)
28.1 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

FileName: IDA13B.WK1

1988 Operational Releases

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surfac	Red Marrow
Immersion	1.14E-02	3.33E-01	1.26E-02	1.01E-02	9.52E-03	1.11E-02	1.02E-02
Inhalation	1.60E-03	2.71E-04	3.35E-03	1.02E-02	3.56E-04	4.74E-04	3.07E-04
Grnd Surface	6.80E-02	1.41E-01	7.30E-02	5.60E-02	5.16E-02	6.81E-02	5.49E-02
Ingestion	5.60E-02	4.64E-03	1.73E+00	4.73E-03	4.64E-03	4.98E-03	4.84E-03
TOTALS	1.37E-01	4.79E-01	1.82E+00	8.11E-02	6.61E-02	8.46E-02	7.02E-02

Table C-24. (continued).

Report Date: 15-Apr-91
Infant receptor

1988 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	7.0E+02	8.20E-13	7.19E-09	0.00E+00
C-14	5.02E+07	2.7E+00	3.16E-15	2.77E-11	0.00E+00
Na-24	1.50E+01	8.9E-03	9.54E-18	8.36E-14	1.80E-03
Ar-41	1.83E+00	2.1E+03	1.19E-12	1.04E-08	0.00E+00
Cr-51	6.65E+02	1.0E-02	1.17E-17	1.02E-13	1.80E-03
Br-82	3.53E+01	2.2E-02	2.48E-17	2.18E-13	1.80E-03
Kr-85	9.40E+04	1.7E+05	1.99E-10	1.75E-06	0.00E+00
Kr-85m	4.48E+00	3.8E+01	3.31E-14	2.90E-10	0.00E+00
Kr-87	1.27E+00	7.7E+01	3.17E-14	2.77E-10	0.00E+00
Kr-88	2.84E+00	7.4E+01	5.42E-14	4.75E-10	0.00E+00
Rb-89	2.57E-01	8.2E-01	5.43E-18	4.76E-14	1.80E-03
Sr-90	2.51E+05	3.4E-04	3.98E-19	3.49E-15	1.80E-03
Y-91m	8.29E-01	1.8E-03	4.22E-19	3.70E-15	1.80E-03
Tc-99m	6.02E+00	3.7E-03	3.47E-18	3.04E-14	1.80E-03
Ru-106	8.84E+03	1.9E-01	2.22E-16	1.95E-12	1.80E-03
Sb-125	2.43E+04	7.4E+00	8.67E-15	7.60E-11	1.80E-03
I-129 (ELE)	1.38E+11	7.3E-02	8.55E-17	7.49E-13	1.80E-02
I-129 (ORG)	1.38E+11	1.5E-01	1.76E-16	1.54E-12	1.80E-04
I-131	1.93E+02	1.7E-04	1.98E-19	1.73E-15	1.80E-02
I-132	2.30E+00	2.0E-03	1.31E-18	1.15E-14	1.80E-02
I-133	2.08E+01	5.8E-04	6.37E-19	5.59E-15	1.80E-02
Xe-133	1.26E+02	4.5E+02	5.22E-13	4.57E-09	0.00E+00
Xe-135	9.11E+00	1.9E+02	1.92E-13	1.69E-09	0.00E+00
Xe-135m	2.56E-01	4.3E+01	2.77E-16	2.43E-12	0.00E+00
Xe-138	2.36E-01	2.0E+02	8.20E-16	7.19E-12	0.00E+00
Cs-137	2.65E+05	6.8E-04	7.96E-19	6.98E-15	1.80E-03
Cs-138	5.37E-01	8.3E-01	8.13E-17	7.13E-13	1.80E-03
Ba-139	1.39E+00	8.4E-02	3.76E-17	3.30E-13	1.80E-03
U-238	3.91E+13	1.3E-05	1.52E-20	1.33E-16	1.80E-03
Pu-238	7.69E+05	1.4E-05	1.64E-20	1.44E-16	1.80E-03
Pu-239/240	2.12E+08	2.1E-06	2.46E-21	2.16E-17	1.80E-03
TOTAL		1.74E+05			

Table C-24. (continued).

Report Date: 15-Apr-91
Infant receptor

1988 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	3.10E-03	3.09E-03	3.09E-03	3.09E-03	3.09E-03	3.09E-03	3.09E-03
C-14	1.65E-03	1.65E-03	1.65E-03	1.65E-03	1.65E-03	1.65E-03	1.65E-03
Na-24	5.78E-07	1.27E-06	5.95E-07	5.60E-07	4.75E-07	5.27E-07	5.24E-07
Ar-41	7.86E-03	1.45E-02	8.70E-03	6.98E-03	6.60E-03	7.46E-03	7.04E-03
Cr-51	1.90E-07	2.33E-07	1.89E-07	1.81E-07	1.38E-07	1.90E-07	1.57E-07
Br-82	1.95E-06	2.51E-06	2.02E-06	1.89E-06	1.51E-06	1.90E-06	1.73E-06
Kr-85	2.23E-03	3.14E-01	2.41E-03	1.94E-03	1.81E-03	2.27E-03	1.98E-03
Kr-85m	2.70E-05	8.66E-05	2.98E-05	2.22E-05	2.04E-05	3.27E-05	2.13E-05
Kr-87	1.41E-04	5.22E-04	1.51E-04	1.25E-04	1.19E-04	1.36E-04	1.28E-04
Kr-88	8.30E-04	2.27E-03	8.91E-04	7.42E-04	7.05E-04	7.84E-04	7.55E-04
Rb-89	6.38E-08	1.37E-07	6.92E-08	5.82E-08	5.34E-08	6.02E-08	5.74E-08
Sr-90	2.12E-05	4.94E-04	1.80E-06	1.82E-06	1.80E-06	1.63E-04	1.16E-04
Y-91m	1.27E-09	1.84E-09	1.37E-09	1.25E-09	1.02E-09	1.27E-09	1.12E-09
Tc-99m	4.48E-09	5.40E-09	5.68E-09	3.90E-09	3.17E-09	5.48E-09	3.26E-09
Ru-106	1.57E-03	2.01E-02	3.60E-04	9.19E-03	2.90E-04	3.48E-04	3.41E-04
Sb-125	6.41E-02	1.11E-01	6.89E-02	5.49E-02	4.98E-02	6.45E-02	5.39E-02
I-129 (ELE)	5.40E-02	9.16E-03	1.69E+00	2.03E-03	1.54E-03	3.47E-03	8.45E-04
I-129 (ORG)	1.17E-03	1.88E-04	3.69E-02	4.40E-05	3.18E-05	7.16E-05	1.76E-05
I-131	5.79E-06	2.90E-07	1.92E-04	1.39E-07	1.04E-07	1.41E-07	1.26E-07
I-132	6.76E-08	1.76E-07	1.88E-07	7.24E-08	5.11E-08	6.03E-08	5.52E-08
I-133	7.75E-08	3.83E-07	8.15E-07	4.83E-08	4.51E-08	5.55E-08	4.93E-08
Xe-133	9.17E-05	3.03E-04	1.05E-04	6.77E-05	6.05E-05	1.20E-04	4.42E-05
Xe-135	2.40E-04	6.99E-04	2.61E-04	2.02E-04	1.87E-04	2.69E-04	2.06E-04
Xe-135m	5.96E-07	9.89E-07	6.46E-07	5.18E-07	4.82E-07	6.04E-07	5.29E-07
Xe-138	5.14E-06	1.06E-05	5.53E-06	4.56E-06	4.32E-06	4.96E-06	4.62E-06
Cs-137	6.08E-05	1.79E-04	6.59E-05	5.35E-05	5.06E-05	6.07E-05	5.71E-05
Cs-138	1.10E-06	2.62E-06	1.20E-06	1.04E-06	9.21E-07	1.04E-06	9.89E-07
Ba-139	1.64E-08	5.96E-07	8.94E-09	5.51E-08	6.14E-09	9.57E-09	6.41E-09
U-238	2.71E-06	2.74E-05	7.43E-08	2.11E-05	5.00E-08	1.88E-06	1.70E-07
Pu-238	1.56E-05	6.53E-07	6.34E-07	1.60E-05	7.10E-05	1.73E-04	2.26E-05
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-24. (continued).

Report Date: 15-Apr-91
Infant receptor

1988 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	9.20E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.28E-07	3.34E-07	2.42E-07	2.03E-07	1.93E-07	2.12E-07	2.07E-07
Ar-41	7.86E-03	1.45E-02	8.70E-03	6.98E-03	6.60E-03	7.46E-03	7.04E-03
Cr-51	1.85E-09	2.39E-09	2.00E-09	1.58E-09	1.46E-09	2.01E-09	1.62E-09
Br-82	3.40E-07	4.71E-07	3.74E-07	3.00E-07	2.80E-07	3.32E-07	3.03E-07
Kr-85	2.23E-03	3.14E-01	2.41E-03	1.94E-03	1.81E-03	2.27E-03	1.98E-03
Kr-85m	2.70E-05	8.66E-05	2.98E-05	2.22E-05	2.04E-05	3.27E-05	2.13E-05
Kr-87	1.41E-04	5.22E-04	1.51E-04	1.25E-04	1.19E-04	1.36E-04	1.28E-04
Kr-88+D	8.30E-04	2.27E-03	8.91E-04	7.42E-04	7.05E-04	7.84E-04	7.55E-04
Rb-89	6.18E-08	1.25E-07	6.73E-08	5.48E-08	5.20E-08	5.86E-08	5.59E-08
Sr-90+D	0.00E+00	3.38E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.13E-09	1.60E-09	1.23E-09	9.88E-10	9.20E-10	1.14E-09	1.01E-09
Tc-99m	2.30E-09	2.92E-09	2.57E-09	1.87E-09	1.69E-09	2.93E-09	1.70E-09
Ru-106+D	2.33E-07	2.85E-06	2.56E-07	2.05E-07	1.91E-07	2.33E-07	2.09E-07
Sb-125+D	1.89E-05	2.97E-05	2.05E-05	1.62E-05	1.50E-05	1.92E-05	1.64E-05
I-129 (ELE)	3.56E-09	9.91E-09	3.55E-09	1.76E-09	1.41E-09	3.09E-09	6.68E-10
I-129 (ORG)	7.32E-09	2.04E-08	7.29E-09	3.62E-09	2.90E-09	6.36E-09	1.37E-09
I-131	3.77E-10	6.96E-10	4.09E-10	3.24E-10	3.00E-10	3.99E-10	3.34E-10
I-132	1.56E-08	2.56E-08	1.72E-08	1.38E-08	1.29E-08	1.52E-08	1.39E-08
I-133	1.95E-09	4.44E-09	2.13E-09	1.71E-09	1.60E-09	1.96E-09	1.74E-09
Xe-133	9.17E-05	3.03E-04	1.05E-04	6.77E-05	6.05E-05	1.20E-04	4.42E-05
Xe-135	2.40E-04	6.99E-04	2.61E-04	2.02E-04	1.87E-04	2.69E-04	2.06E-04
Xe-135m	5.96E-07	9.89E-07	6.46E-07	5.18E-07	4.82E-07	6.04E-07	5.29E-07
Xe-138	5.14E-06	1.06E-05	5.53E-06	4.56E-06	4.32E-06	4.96E-06	4.62E-06
Cs-137+D	2.30E-09	4.16E-09	2.54E-09	2.02E-09	1.89E-09	2.29E-09	2.06E-09
Cs-138	1.03E-06	2.16E-06	1.13E-06	9.18E-07	8.69E-07	9.83E-07	9.34E-07
Ba-139	6.77E-09	2.71E-07	7.48E-09	5.60E-09	5.15E-09	8.01E-09	5.37E-09
U-238+D	1.51E-12	1.51E-12	1.00E-10	1.70E-12	1.26E-12	1.17E-12	1.70E-12
Pu-238	7.23E-15	5.01E-14	2.80E-15	1.93E-15	1.17E-15	3.42E-15	8.70E-16
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	1.14E-02	3.33E-01	1.26E-02	1.01E-02	9.52E-03	1.11E-02	1.02E-02

Table C-24. (continued).

Report Date: 15-Apr-91
Infant receptor

1988 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	2.32E-04	2.32E-04	2.32E-04	2.32E-04	2.32E-04	2.32E-04	2.32E-04
C-14	2.11E-05	2.11E-05	2.11E-05	2.11E-05	2.11E-05	2.11E-05	2.11E-05
Na-24	1.26E-08	0.00E+00	0.00E+00	5.81E-08	0.00E+00	0.00E+00	1.03E-08
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	4.21E-09	0.00E+00	0.00E+00	3.24E-08	0.00E+00	0.00E+00	0.00E+00
Br-82	4.47E-08	0.00E+00	0.00E+00	2.17E-07	0.00E+00	4.81E-08	2.78E-08
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.78E-10	0.00E+00	0.00E+00	1.88E-09	0.00E+00	0.00E+00	0.00E+00
Sr-90	2.45E-07	2.04E-08	2.04E-08	4.69E-08	2.04E-08	1.94E-06	1.39E-06
Y-91m	1.81E-11	0.00E+00	0.00E+00	1.52E-10	0.00E+00	0.00E+00	0.00E+00
Tc-99m	1.54E-10	0.00E+00	8.66E-10	4.09E-10	0.00E+00	0.00E+00	6.25E-11
Ru-106	1.12E-03	1.71E-05	1.94E-05	8.91E-03	2.28E-05	2.28E-05	2.71E-05
Sb-125	1.18E-04	0.00E+00	0.00E+00	9.72E-04	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	3.07E-05	6.13E-08	1.01E-03	1.14E-06	6.13E-08	1.67E-07	9.64E-08
I-129 (ORG)	6.30E-05	1.26E-07	2.07E-03	2.34E-06	1.26E-07	3.43E-07	1.98E-07
I-131	6.98E-08	3.04E-10	2.33E-06	7.09E-09	3.34E-10	4.46E-10	4.57E-10
I-132	6.73E-09	5.31E-10	1.21E-07	1.89E-08	9.42E-10	8.07E-10	8.07E-10
I-133	4.77E-09	0.00E+00	1.59E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	5.31E-08	4.04E-08	4.90E-08	8.16E-08	4.90E-08	5.31E-08	6.94E-08
Cs-138	9.91E-09	0.00E+00	0.00E+00	6.64E-08	0.00E+00	0.00E+00	0.00E+00
Ba-139	8.33E-09	0.00E+00	0.00E+00	4.84E-08	0.00E+00	0.00E+00	0.00E+00
U-238	2.53E-06	0.00E+00	0.00E+00	2.11E-05	0.00E+00	0.00E+00	0.00E+00
Pu-238	1.51E-05	6.13E-07	6.13E-07	1.60E-05	6.88E-05	1.68E-04	2.18E-05
Pu-239/240	2.39E-06	9.06E-08	9.06E-08	2.27E-06	1.14E-05	2.77E-05	3.40E-06
TOTALS	1.60E-03	2.71E-04	3.35E-03	1.02E-02	3.56E-04	4.74E-04	3.07E-04

Table C-24. (continued).

Report Date: 15-Apr-91
Infant receptor

1988 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	3.32E-07	9.34E-07	3.53E-07	2.95E-07	2.82E-07	3.09E-07	3.02E-07
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.73E-07	2.30E-07	1.87E-07	1.47E-07	1.36E-07	1.88E-07	1.52E-07
Br-82	1.49E-06	2.04E-06	1.64E-06	1.31E-06	1.23E-06	1.46E-06	1.33E-06
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.74E-09	1.21E-08	1.90E-09	1.54E-09	1.46E-09	1.65E-09	1.57E-09
Sr-90+D	0.00E+00	4.92E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.25E-10	2.36E-10	1.36E-10	1.09E-10	1.02E-10	1.26E-10	1.11E-10
Tc-99m	2.02E-09	2.49E-09	2.24E-09	1.63E-09	1.48E-09	2.56E-09	1.49E-09
Ru-106+D	2.70E-04	2.01E-02	2.95E-04	2.37E-04	2.21E-04	2.71E-04	2.42E-04
Sb-125+D	6.39E-02	1.11E-01	6.89E-02	5.39E-02	4.98E-02	6.45E-02	5.39E-02
I-129 (ELE)	3.77E-03	9.06E-03	3.72E-03	1.84E-03	1.45E-03	3.21E-03	6.90E-04
I-129 (ORG)	7.76E-05	1.86E-04	7.65E-05	3.77E-05	2.97E-05	6.59E-05	1.42E-05
I-131	1.01E-07	2.65E-07	1.09E-07	8.66E-08	8.04E-08	1.07E-07	8.91E-08
I-132	4.52E-08	1.50E-07	4.99E-08	3.97E-08	3.73E-08	4.42E-08	4.05E-08
I-133	5.32E-08	3.78E-07	5.81E-08	4.66E-08	4.35E-08	5.36E-08	4.76E-08
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	5.47E-05	1.74E-04	6.01E-05	4.80E-05	4.47E-05	5.42E-05	4.89E-05
Cs-138	6.14E-08	4.57E-07	6.68E-08	5.45E-08	5.17E-08	5.85E-08	5.51E-08
Ba-139	1.32E-09	3.26E-07	1.46E-09	1.09E-09	9.95E-10	1.57E-09	1.04E-09
U-238+D	6.89E-08	2.74E-05	7.42E-08	5.46E-08	5.00E-08	7.63E-08	4.78E-08
Pu-238	2.33E-09	1.95E-08	4.10E-10	3.31E-10	7.14E-11	5.56E-10	1.22E-10
Pu-239/240	1.86E-10	1.38E-09	5.72E-11	4.41E-11	2.39E-11	7.25E-11	2.75E-11
TOTALS	6.80E-02	1.41E-01	7.30E-02	5.60E-02	5.16E-02	6.81E-02	5.49E-02

Table C-24. (continued).

Report Date: 15-Apr-91
Infant receptor

1988 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	2.87E-03	2.86E-03	2.86E-03	2.86E-03	2.86E-03	2.86E-03	2.86E-03
C-14	1.63E-03	1.63E-03	1.63E-03	1.63E-03	1.63E-03	1.63E-03	1.63E-03
Na-24	4.99E-09	0.00E+00	0.00E+00	3.42E-09	0.00E+00	6.06E-09	4.99E-09
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.10E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.73E-09
Br-82	7.22E-08	0.00E+00	0.00E+00	5.95E-08	0.00E+00	5.95E-08	6.37E-08
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	8.92E-42	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	2.10E-05	1.78E-06	1.78E-06	1.78E-06	1.78E-06	1.61E-04	1.15E-04
Y-91m	2.44E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	4.24E-13	0.00E+00	2.19E-12	0.00E+00	0.00E+00	0.00E+00	1.62E-13
Ru-106	1.78E-04	4.40E-05	4.60E-05	4.60E-05	4.60E-05	5.40E-05	7.19E-05
Sb-125	7.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	5.02E-02	1.00E-04	1.69E+00	1.91E-04	9.58E-05	2.60E-04	1.55E-04
I-129 (ORG)	1.03E-03	2.06E-06	3.47E-02	3.93E-06	1.97E-06	5.35E-06	3.19E-06
I-131	5.62E-06	2.40E-08	1.89E-04	4.54E-08	2.35E-08	3.37E-08	3.58E-08
I-132	1.42E-13	1.19E-14	3.25E-12	2.37E-14	1.90E-14	1.66E-14	1.90E-14
I-133	1.75E-08	0.00E+00	5.96E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	6.07E-06	4.86E-06	5.77E-06	5.46E-06	5.77E-06	6.37E-06	8.19E-06
Cs-138	2.27E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	1.15E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-238	1.12E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.81E-06	1.22E-07
Pu-238	4.34E-07	2.00E-08	2.00E-08	2.04E-08	2.17E-06	5.34E-06	7.34E-07
Pu-239/240	7.02E-08	3.01E-09	3.01E-09	3.01E-09	3.56E-07	9.02E-07	1.10E-07
TOTALS	5.60E-02	4.64E-03	1.73E+00	4.73E-03	4.64E-03	4.98E-03	4.84E-03

Table C-25. Dose evaluation for 1971 operational releases (adult)

Report Date: 15-Apr-91
FileName: IDA13A.WK1

Operational Releases

Calendar Year..... 1971
Receptor Location..... Atomic City
Receptor Designation..... Adult
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

5.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
19.1 : Distance to Receptor (km)
16.4 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
9.59E-01 : Breathing Rate (m**3/hr) 1.38E-87
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

6.08E+03

Report Date: 15-Apr-91

1971 Operational Releases

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	5.16E-02	2.33E-01	5.69E-02	4.59E-02	4.34E-02	4.91E-02	4.64E-02
Inhalation	3.38E-01	1.14E-03	5.96E-02	2.15E+00	2.99E-01	1.35E+00	1.16E-01
Grnd Surface	7.90E-01	1.71E+01	8.69E-01	6.93E-01	6.46E-01	7.85E-01	7.05E-01
Ingestion	6.40E-01	1.08E-02	1.99E+00	1.76E-01	1.68E-02	4.76E+00	2.18E+00
TOTALS	1.82E+00	1.73E+01	2.98E+00	3.06E+00	1.01E+00	6.95E+00	3.05E+00

Table C-25. (continued).

Report Date: 15-Apr-91
Adult receptor

1971 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	3.3E+03	2.15E-12	1.88E-08	0.00E+00
Ar-41	1.83E+00	1.6E+04	6.69E-12	5.87E-08	0.00E+00
Co-60	4.62E+04	5.1E-03	3.32E-18	2.91E-14	1.80E-03
Kr-85	9.40E+04	1.3E+05	8.46E-11	7.41E-07	0.00E+00
Kr-87	1.27E+00	1.3E+00	4.48E-16	3.93E-12	0.00E+00
Kr-88	2.84E+00	8.0E+02	3.92E-13	3.43E-09	0.00E+00
Kr-89	5.27E-02	3.6E+02	5.18E-20	4.54E-16	0.00E+00
Rb-89	2.57E-01	1.6E+01	4.52E-16	3.96E-12	1.80E-03
Sr-90	2.51E+05	1.4E+01	9.11E-15	7.99E-11	1.80E-03
Zr-95	1.54E+03	1.7E+00	1.11E-15	9.69E-12	1.80E-03
Ru-103	9.44E+02	1.8E-01	1.17E-16	1.03E-12	1.80E-03
Ru-106	8.84E+03	4.0E+01	2.60E-14	2.28E-10	1.80E-03
Sb-125	2.43E+04	1.7E+00	1.11E-15	9.70E-12	1.80E-03
I-129 (ELE)	1.38E+11	3.2E-02	2.08E-17	1.83E-13	1.80E-02
I-129 (ORG)	1.38E+11	6.4E-02	4.16E-17	3.65E-13	1.80E-04
I-131	1.93E+02	8.6E+00	5.59E-15	4.90E-11	1.80E-02
Xe-133	1.26E+02	4.2E+00	2.72E-15	2.38E-11	0.00E+00
Xe-135	9.11E+00	2.7E+00	1.61E-15	1.41E-11	0.00E+00
Xe-138	2.36E-01	1.4E+03	2.96E-14	2.59E-10	0.00E+00
Cs-134	1.81E+04	5.7E+00	3.71E-15	3.25E-11	1.80E-03
Cs-137	2.65E+05	1.5E+01	9.76E-15	8.56E-11	1.80E-03
Cs-138	5.37E-01	2.6E+01	3.76E-15	3.30E-11	1.80E-03
Ce-144	6.82E+03	3.1E+01	2.02E-14	1.77E-10	1.80E-03
Pu-238	7.69E+05	2.6E-02	1.69E-17	1.48E-13	1.80E-03
Pu-239/240	2.12E+08	3.8E-03	2.47E-18	2.17E-14	1.80E-03
TOTAL		1.52E+05			

Table C-25. (continued).

Report Date: 15-Apr-91
Adult receptor

1971 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.19E-02	1.19E-02	1.19E-02	1.19E-02	1.19E-02	0.00E+00	1.19E-02
Ar-41	4.43E-02	8.16E-02	4.91E-02	3.94E-02	3.73E-02	4.21E-02	3.97E-02
Co-60	2.54E-04	3.06E-04	2.52E-04	2.55E-04	2.32E-04	2.16E-04	2.21E-04
Kr-85	9.47E-04	1.34E-01	1.02E-03	8.24E-04	7.68E-04	9.64E-04	8.43E-04
Kr-87	2.00E-06	7.40E-06	2.14E-06	1.77E-06	1.68E-06	1.92E-06	1.81E-06
Kr-88	5.99E-03	1.64E-02	6.43E-03	5.36E-03	5.10E-03	5.67E-03	5.45E-03
Kr-89	5.28E-10	1.26E-09	5.64E-10	4.66E-10	4.43E-10	5.02E-10	4.76E-10
Rb-89	5.44E-06	1.14E-05	5.76E-06	5.64E-06	4.45E-06	5.02E-06	4.78E-06
Sr-90	4.71E-01	1.13E+01	0.00E+00	8.42E-01	0.00E+00	4.58E+00	2.00E+00
Zr-95	3.00E-03	3.75E-03	3.03E-03	2.41E-03	2.25E-03	6.12E-03	2.89E-03
Ru-103	4.97E-05	5.36E-05	3.95E-05	8.98E-05	2.95E-05	3.72E-05	3.24E-05
Ru-106	1.45E-01	2.35E+00	3.45E-02	8.37E-01	2.59E-02	3.18E-02	2.83E-02
Sb-125	8.34E-03	1.42E-02	8.79E-03	7.63E-03	6.36E-03	8.23E-03	6.88E-03
I-129 (ELE)	1.45E-02	2.21E-03	4.54E-01	4.47E-04	3.52E-04	7.81E-04	1.68E-04
I-129 (ORG)	3.53E-04	4.41E-05	1.11E-02	8.94E-06	7.05E-06	1.56E-05	3.36E-06
I-131	4.43E-02	7.50E-03	1.41E+00	2.46E-03	2.28E-03	3.03E-03	2.53E-03
Xe-133	4.78E-07	1.58E-06	5.45E-07	3.53E-07	3.15E-07	6.24E-07	2.30E-07
Xe-135	2.01E-06	5.85E-06	2.19E-06	1.69E-06	1.56E-06	2.25E-06	1.72E-06
Xe-138	1.85E-04	3.81E-04	2.00E-04	1.64E-04	1.56E-04	1.79E-04	1.67E-04
Cs-134	1.26E-01	1.21E-01	1.27E-01	1.12E-01	5.70E-02	1.16E-01	1.15E-01
Cs-137	7.93E-01	2.13E+00	8.55E-01	7.06E-01	5.48E-01	7.82E-01	7.17E-01
Cs-138	5.33E-05	1.21E-04	5.53E-05	6.36E-05	4.26E-05	4.82E-05	4.58E-05
Ce-144	7.32E-02	1.17E+00	5.05E-03	4.96E-01	3.59E-03	4.97E-03	3.64E-03
Pu-238	6.67E-02	2.01E-05	4.23E-07	3.42E-07	2.61E-01	1.17E+00	9.72E-02
Pu-239/240	1.08E-02	1.39E-06	5.75E-08	4.43E-08	4.24E-02	1.97E-01	1.57E-02

Table C-25. (continued).

Report Date: 15-Apr-91
Adult receptor

1971 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ar-41	4.43E-02	8.16E-02	4.91E-02	3.94E-02	3.73E-02	4.21E-02	3.97E-02
Co-60	4.31E-08	5.87E-08	4.78E-08	3.81E-08	3.61E-08	4.08E-08	3.85E-08
Kr-85	9.47E-04	1.34E-01	1.02E-03	8.24E-04	7.68E-04	9.64E-04	8.43E-04
Kr-87	2.00E-06	7.40E-06	2.14E-06	1.77E-06	1.68E-06	1.92E-06	1.81E-06
Kr-88+D	5.99E-03	1.64E-02	6.43E-03	5.36E-03	5.10E-03	5.67E-03	5.45E-03
Kr-89	5.28E-10	1.26E-09	5.64E-10	4.66E-10	4.43E-10	5.02E-10	4.76E-10
Rb-89	5.15E-06	1.04E-05	5.60E-06	4.56E-06	4.33E-06	4.88E-06	4.65E-06
Sr-90+D	0.00E+00	7.73E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95+D	1.38E-05	1.93E-05	1.53E-05	1.22E-05	1.14E-05	1.35E-05	1.24E-05
Ru-103+D	2.83E-07	3.97E-07	3.06E-07	2.47E-07	2.29E-07	2.89E-07	2.53E-07
Ru-106+D	2.73E-05	3.33E-04	2.99E-05	2.40E-05	2.24E-05	2.73E-05	2.45E-05
Sb-125+D	2.41E-06	3.79E-06	2.61E-06	2.07E-06	1.92E-06	2.45E-06	2.09E-06
I-129 (ELE)	8.68E-10	2.41E-09	8.64E-10	4.29E-10	3.43E-10	7.53E-10	1.63E-10
I-129 (ORG)	1.74E-09	4.83E-09	1.73E-09	8.57E-10	6.87E-10	1.51E-09	3.25E-10
I-131	1.07E-05	1.97E-05	1.16E-05	9.16E-06	8.49E-06	1.13E-05	9.44E-06
Xe-133	4.78E-07	1.58E-06	5.45E-07	3.53E-07	3.15E-07	6.24E-07	2.30E-07
Xe-135	2.01E-06	5.85E-06	2.19E-06	1.69E-06	1.56E-06	2.25E-06	1.72E-06
Xe-138	1.85E-04	3.81E-04	2.00E-04	1.64E-04	1.56E-04	1.79E-04	1.67E-04
Cs-134	2.97E-05	4.34E-05	3.27E-05	2.61E-05	2.44E-05	2.93E-05	2.66E-05
Cs-137+D	2.82E-05	5.10E-05	3.11E-05	2.47E-05	2.32E-05	2.80E-05	2.53E-05
Cs-138	4.77E-05	1.00E-04	5.22E-05	4.25E-05	4.02E-05	4.55E-05	4.32E-05
Ce-144+D	5.39E-06	2.03E-04	5.89E-06	4.60E-06	4.31E-06	5.72E-06	4.44E-06
Pu-238	7.46E-12	5.17E-11	2.89E-12	1.99E-12	1.21E-12	3.53E-12	8.98E-13
Pu-239/240	1.02E-12	3.58E-12	8.38E-13	5.98E-13	5.07E-13	9.91E-13	4.82E-13
TOTALS	5.16E-02	2.33E-01	5.69E-02	4.59E-02	4.34E-02	4.91E-02	4.64E-02

Table C-25. (continued).

Report Date: 15-Apr-91
Adult receptor

1971 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.14E-03	1.14E-03	1.14E-03	1.14E-03	1.14E-03	0.00E+00	1.14E-03
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	4.18E-06	0.00E+00	0.00E+00	3.63E-05	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.41E-07	0.00E+00	0.00E+00	9.50E-07	0.00E+00	0.00E+00	0.00E+00
Sr-90	9.95E-02	0.00E+00	0.00E+00	8.42E-01	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.77E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.44E-03	4.46E-04
Ru-103	7.67E-06	0.00E+00	0.00E+00	5.80E-05	0.00E+00	0.00E+00	0.00E+00
Ru-106	9.63E-02	0.00E+00	0.00E+00	8.09E-01	0.00E+00	0.00E+00	0.00E+00
Sb-125	9.11E-05	0.00E+00	0.00E+00	7.53E-04	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	3.15E-05	0.00E+00	1.03E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	6.30E-05	0.00E+00	2.07E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	1.50E-03	0.00E+00	5.17E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.47E-03	0.00E+00	1.28E-03	1.37E-03	0.00E+00	1.28E-03	1.37E-03
Cs-137	2.63E-03	0.00E+00	2.38E-03	2.71E-03	0.00E+00	2.38E-03	2.54E-03
Cs-138	2.78E-06	0.00E+00	0.00E+00	1.86E-05	0.00E+00	0.00E+00	0.00E+00
Ce-144	5.93E-02	0.00E+00	0.00E+00	4.92E-01	0.00E+00	0.00E+00	0.00E+00
Pu-238	6.54E-02	0.00E+00	0.00E+00	0.00E+00	2.56E-01	1.15E+00	9.53E-02
Pu-239/240	1.06E-02	0.00E+00	0.00E+00	0.00E+00	4.16E-02	1.93E-01	1.54E-02
TOTALS	3.38E-01	1.14E-03	5.96E-02	2.15E+00	2.99E-01	1.35E+00	1.16E-01

Table C-25. (continued).

Report Date: 15-Apr-91
Adult receptor

1971 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	2.27E-04	3.06E-04	2.52E-04	2.02E-04	1.91E-04	2.16E-04	2.04E-04
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.45E-07	1.00E-06	1.58E-07	1.29E-07	1.22E-07	1.38E-07	1.31E-07
Sr-90+D	0.00E+00	1.13E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95+D	2.72E-03	3.73E-03	3.01E-03	2.39E-03	2.24E-03	2.66E-03	2.43E-03
Ru-103+D	3.62E-05	5.32E-05	3.92E-05	3.15E-05	2.93E-05	3.69E-05	3.22E-05
Ru-106+D	3.16E-02	2.35E+00	3.44E-02	2.77E-02	2.59E-02	3.17E-02	2.83E-02
Sb-125+D	8.15E-03	1.42E-02	8.79E-03	6.88E-03	6.36E-03	8.23E-03	6.88E-03
I-129 (ELE)	9.19E-04	2.21E-03	9.07E-04	4.47E-04	3.52E-04	7.81E-04	1.68E-04
I-129 (ORG)	1.84E-05	4.41E-05	1.81E-05	8.94E-06	7.04E-06	1.56E-05	3.36E-06
I-131	2.85E-03	7.48E-03	3.09E-03	2.45E-03	2.27E-03	3.02E-03	2.52E-03
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	6.93E-02	1.21E-01	7.63E-02	6.10E-02	5.70E-02	6.84E-02	6.18E-02
Cs-137+D	6.70E-01	2.13E+00	7.37E-01	5.88E-01	5.48E-01	6.65E-01	5.99E-01
Cs-138	2.84E-06	2.11E-05	3.09E-06	2.52E-06	2.39E-06	2.71E-06	2.55E-06
Ce-144+D	4.59E-03	1.17E+00	5.05E-03	3.86E-03	3.59E-03	4.97E-03	3.64E-03
Pu-238	2.40E-06	2.01E-05	4.23E-07	3.42E-07	7.36E-08	5.74E-07	1.26E-07
Pu-239/240	1.87E-07	1.39E-06	5.75E-08	4.43E-08	2.40E-08	7.29E-08	2.77E-08
TOTALS	7.90E-01	1.71E+01	8.69E-01	6.93E-01	6.46E-01	7.85E-01	7.05E-01

Table C-25. (continued).

Report Date: 15-Apr-91
Adult receptor

1971 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.08E-02	1.08E-02	1.08E-02	1.08E-02	1.08E-02	0.00E+00	1.08E-02
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	2.20E-05	0.00E+00	0.00E+00	1.61E-05	4.06E-05	0.00E+00	1.69E-05
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.75E-38	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	3.72E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.58E+00	2.00E+00
Zr-95	8.86E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	5.56E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.75E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-125	9.53E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.36E-02	0.00E+00	4.52E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	2.72E-04	0.00E+00	9.03E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	3.99E-02	0.00E+00	1.36E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	5.48E-02	0.00E+00	4.96E-02	4.96E-02	0.00E+00	4.67E-02	5.19E-02
Cs-137	1.20E-01	0.00E+00	1.15E-01	1.15E-01	0.00E+00	1.15E-01	1.15E-01
Cs-138	6.71E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	9.24E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu-238	1.31E-03	0.00E+00	0.00E+00	0.00E+00	5.17E-03	2.31E-02	1.93E-03
Pu-239/240	2.17E-04	0.00E+00	0.00E+00	0.00E+00	8.06E-04	3.93E-03	2.97E-04
TOTALS	6.40E-01	1.08E-02	1.99E+00	1.76E-01	1.68E-02	4.76E+00	2.18E+00

Table C-26. Dose evaluation for 1971 operational releases (infant)

Report Date: 15-Apr-91
FileName: IDA13B.WK1

Operational Releases

Calendar Year..... 1971
Receptor Location..... Atomic City
Receptor Designation..... Infant
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

5.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
19.1 : Distance to Receptor (km)
16.4 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

FileName: IDA13B.WK1

1971 Operational Releases

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surfac	Red Marrow
Immersion	5.16E-02	2.33E-01	5.69E-02	4.59E-02	4.34E-02	4.91E-02	4.64E-02
Inhalation	2.40E-01	4.80E-03	7.20E-02	1.71E+00	9.28E-02	2.60E-01	7.41E-02
Grnd Surface	7.90E-01	1.71E+01	8.69E-01	6.93E-01	6.46E-01	7.85E-01	7.05E-01
Ingestion	7.91E-01	1.34E-01	5.92E+00	1.48E-01	1.59E-01	3.83E+00	2.78E+00
TOTALS	1.87E+00	1.74E+01	6.92E+00	2.59E+00	9.41E-01	4.92E+00	3.61E+00

Table C-26. (continued).

Report Date: 15-Apr-91
Infant receptor

1971 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	3.3E+03	2.15E-12	1.88E-08	0.00E+00
Ar-41	1.83E+00	1.6E+04	6.69E-12	5.87E-08	0.00E+00
Co-60	4.62E+04	5.1E-03	3.32E-18	2.91E-14	1.80E-03
Kr-85	9.40E+04	1.3E+05	8.46E-11	7.41E-07	0.00E+00
Kr-87	1.27E+00	1.3E+00	4.48E-16	3.93E-12	0.00E+00
Kr-88	2.84E+00	8.0E+02	3.92E-13	3.43E-09	0.00E+00
Kr-89	5.27E-02	3.6E+02	5.18E-20	4.54E-16	0.00E+00
Rb-89	2.57E-01	1.6E+01	4.52E-16	3.96E-12	1.80E-03
Sr-90	2.51E+05	1.4E+01	9.11E-15	7.99E-11	1.80E-03
Zr-95	1.54E+03	1.7E+00	1.11E-15	9.69E-12	1.80E-03
Ru-103	9.44E+02	1.8E-01	1.17E-16	1.03E-12	1.80E-03
Ru-106	8.84E+03	4.0E+01	2.60E-14	2.28E-10	1.80E-03
Sb-125	2.43E+04	1.7E+00	1.11E-15	9.70E-12	1.80E-03
I-129 (ELE)	1.38E+11	3.2E-02	2.08E-17	1.83E-13	1.80E-02
I-129 (ORG)	1.38E+11	6.4E-02	4.16E-17	3.65E-13	1.80E-04
I-131	1.93E+02	8.6E+00	5.59E-15	4.90E-11	1.80E-02
Xe-133	1.26E+02	4.2E+00	2.72E-15	2.38E-11	0.00E+00
Xe-135	9.11E+00	2.7E+00	1.61E-15	1.41E-11	0.00E+00
Xe-138	2.36E-01	1.4E+03	2.96E-14	2.59E-10	0.00E+00
Cs-134	1.81E+04	5.7E+00	3.71E-15	3.25E-11	1.80E-03
Cs-137	2.65E+05	1.5E+01	9.76E-15	8.56E-11	1.80E-03
Cs-138	5.37E-01	2.6E+01	3.76E-15	3.30E-11	1.80E-03
Ce-144	6.82E+03	3.1E+01	2.02E-14	1.77E-10	1.80E-03
Pu-238	7.69E+05	2.6E-02	1.69E-17	1.48E-13	1.80E-03
Pu-239/240	2.12E+08	3.8E-03	2.47E-18	2.17E-14	1.80E-03
TOTAL		1.52E+05			

Table C-26. (continued).

Report Date: 15-Apr-91
Infant receptor

1971 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	8.12E-03	8.09E-03	8.09E-03	8.09E-03	8.09E-03	8.09E-03	8.09E-03
Ar-41	4.43E-02	8.16E-02	4.91E-02	3.94E-02	3.73E-02	4.21E-02	3.97E-02
Co-60	2.33E-04	3.06E-04	2.52E-04	2.12E-04	2.00E-04	2.16E-04	2.08E-04
Kr-85	9.47E-04	1.34E-01	1.02E-03	8.24E-04	7.68E-04	9.64E-04	8.43E-04
Kr-87	2.00E-06	7.40E-06	2.14E-06	1.77E-06	1.68E-06	1.92E-06	1.81E-06
Kr-88	5.99E-03	1.64E-02	6.43E-03	5.36E-03	5.10E-03	5.67E-03	5.45E-03
Kr-89	5.28E-10	1.26E-09	5.64E-10	4.66E-10	4.43E-10	5.02E-10	4.76E-10
Rb-89	5.32E-06	1.14E-05	5.76E-06	4.85E-06	4.45E-06	5.02E-06	4.78E-06
Sr-90	4.86E-01	1.13E+01	4.11E-02	4.17E-02	4.11E-02	3.74E+00	2.66E+00
Zr-95	3.08E-03	3.78E-03	3.07E-03	4.22E-03	2.33E-03	2.96E-03	2.54E-03
Ru-103	5.22E-05	5.47E-05	4.11E-05	1.16E-04	3.21E-05	3.91E-05	3.42E-05
Ru-106	1.83E-01	2.35E+00	4.21E-02	1.07E+00	3.39E-02	4.07E-02	3.99E-02
Sb-125	8.18E-03	1.42E-02	8.79E-03	7.00E-03	6.36E-03	8.23E-03	6.88E-03
I-129 (ELE)	1.31E-02	2.23E-03	4.13E-01	4.94E-04	3.76E-04	8.45E-04	2.06E-04
I-129 (ORG)	2.78E-04	4.46E-05	8.74E-03	1.04E-05	7.54E-06	1.70E-05	4.16E-06
I-131	1.64E-01	8.19E-03	5.42E+00	3.94E-03	2.95E-03	4.00E-03	3.55E-03
Xe-133	4.78E-07	1.58E-06	5.45E-07	3.53E-07	3.15E-07	6.24E-07	2.30E-07
Xe-135	2.01E-06	5.85E-06	2.19E-06	1.69E-06	1.56E-06	2.25E-06	1.72E-06
Xe-138	1.85E-04	3.81E-04	2.00E-04	1.64E-04	1.56E-04	1.79E-04	1.67E-04
Cs-134	9.90E-02	1.41E-01	1.06E-01	8.72E-02	8.67E-02	9.93E-02	9.39E-02
Cs-137	7.45E-01	2.19E+00	8.08E-01	6.56E-01	6.20E-01	7.43E-01	7.00E-01
Cs-138	5.10E-05	1.21E-04	5.53E-05	4.81E-05	4.26E-05	4.82E-05	4.58E-05
Ce-144	9.32E-02	1.17E+00	5.51E-03	6.44E-01	1.14E-02	2.10E-02	2.28E-02
Pu-238	1.61E-02	6.73E-04	6.54E-04	1.65E-02	7.32E-02	1.79E-01	2.33E-02
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-26. (continued).

Report Date: 15-Apr-91
Infant receptor

1971 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ar-41	4.43E-02	8.16E-02	4.91E-02	3.94E-02	3.73E-02	4.21E-02	3.97E-02
Co-60	4.31E-08	5.87E-08	4.78E-08	3.81E-08	3.61E-08	4.08E-08	3.85E-08
Kr-85	9.47E-04	1.34E-01	1.02E-03	8.24E-04	7.68E-04	9.64E-04	8.43E-04
Kr-87	2.00E-06	7.40E-06	2.14E-06	1.77E-06	1.68E-06	1.92E-06	1.81E-06
Kr-88+D	5.99E-03	1.64E-02	6.43E-03	5.36E-03	5.10E-03	5.67E-03	5.45E-03
Kr-89	5.28E-10	1.26E-09	5.64E-10	4.66E-10	4.43E-10	5.02E-10	4.76E-10
Rb-89	5.15E-06	1.04E-05	5.60E-06	4.56E-06	4.33E-06	4.88E-06	4.65E-06
Sr-90+D	0.00E+00	7.73E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95+D	1.38E-05	1.93E-05	1.53E-05	1.22E-05	1.14E-05	1.35E-05	1.24E-05
Ru-103+D	2.83E-07	3.97E-07	3.06E-07	2.47E-07	2.29E-07	2.89E-07	2.53E-07
Ru-106+D	2.73E-05	3.33E-04	2.99E-05	2.40E-05	2.24E-05	2.73E-05	2.45E-05
Sb-125+D	2.41E-06	3.79E-06	2.61E-06	2.07E-06	1.92E-06	2.45E-06	2.09E-06
I-129 (ELE)	8.68E-10	2.41E-09	8.64E-10	4.29E-10	3.43E-10	7.53E-10	1.63E-10
I-129 (ORG)	1.74E-09	4.83E-09	1.73E-09	8.57E-10	6.87E-10	1.51E-09	3.25E-10
I-131	1.07E-05	1.97E-05	1.16E-05	9.16E-06	8.49E-06	1.13E-05	9.44E-06
Xe-133	4.78E-07	1.58E-06	5.45E-07	3.53E-07	3.15E-07	6.24E-07	2.30E-07
Xe-135	2.01E-06	5.85E-06	2.19E-06	1.69E-06	1.56E-06	2.25E-06	1.72E-06
Xe-138	1.85E-04	3.81E-04	2.00E-04	1.64E-04	1.56E-04	1.79E-04	1.67E-04
Cs-134	2.97E-05	4.34E-05	3.27E-05	2.61E-05	2.44E-05	2.93E-05	2.66E-05
Cs-137+D	2.82E-05	5.10E-05	3.11E-05	2.47E-05	2.32E-05	2.80E-05	2.53E-05
Cs-138	4.77E-05	1.00E-04	5.22E-05	4.25E-05	4.02E-05	4.55E-05	4.32E-05
Ce-144+D	5.39E-06	2.03E-04	5.89E-06	4.60E-06	4.31E-06	5.72E-06	4.44E-06
Pu-238	7.46E-12	5.17E-11	2.89E-12	1.99E-12	1.21E-12	3.53E-12	8.98E-13
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	5.16E-02	2.33E-01	5.69E-02	4.59E-02	4.34E-02	4.91E-02	4.64E-02

Table C-26. (continued).

Report Date: 15-Apr-91
Infant receptor

1971 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	6.07E-04	6.07E-04	6.07E-04	6.07E-04	6.07E-04	6.07E-04	6.07E-04
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	6.89E-07	0.00E+00	0.00E+00	5.97E-06	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.32E-08	0.00E+00	0.00E+00	1.57E-07	0.00E+00	0.00E+00	0.00E+00
Sr-90	5.60E-03	4.67E-04	4.67E-04	1.07E-03	4.67E-04	4.44E-02	3.18E-02
Zr-95	2.83E-04	2.16E-05	4.26E-05	1.81E-03	7.37E-05	1.42E-04	5.56E-05
Ru-103	1.32E-05	7.80E-07	1.32E-06	8.40E-05	2.11E-06	1.44E-06	1.20E-06
Ru-106	1.31E-01	2.00E-03	2.27E-03	1.04E+00	2.67E-03	2.67E-03	3.17E-03
Sb-125	1.50E-05	0.00E+00	0.00E+00	1.24E-04	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	7.47E-06	1.49E-08	2.45E-04	2.77E-07	1.49E-08	4.07E-08	2.35E-08
I-129 (ORG)	1.49E-05	2.99E-08	4.91E-04	5.55E-07	2.99E-08	8.13E-08	4.69E-08
I-131	1.97E-03	8.59E-06	6.59E-02	2.01E-04	9.45E-06	1.26E-05	1.29E-05
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.23E-04	2.09E-04	3.04E-04	3.99E-04	3.04E-04	3.04E-04	3.23E-04
Cs-137	6.50E-04	4.95E-04	6.00E-04	1.00E-03	6.00E-04	6.50E-04	8.50E-04
Cs-138	4.58E-07	0.00E+00	0.00E+00	3.07E-06	0.00E+00	0.00E+00	0.00E+00
Ce-144	8.16E-02	2.69E-04	3.21E-04	6.40E-01	5.70E-03	9.92E-03	1.14E-02
Pu-238	1.56E-02	6.33E-04	6.33E-04	1.65E-02	7.10E-02	1.73E-01	2.25E-02
Pu-239/240	2.41E-03	9.11E-05	9.11E-05	2.28E-03	1.14E-02	2.79E-02	3.42E-03
TOTALS	2.40E-01	4.80E-03	7.20E-02	1.71E+00	9.28E-02	2.60E-01	7.41E-02

Table C-26. (continued).

Report Date: 15-Apr-91
Infant receptor

1971 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	2.27E-04	3.06E-04	2.52E-04	2.02E-04	1.91E-04	2.16E-04	2.04E-04
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.45E-07	1.00E-06	1.58E-07	1.29E-07	1.22E-07	1.38E-07	1.31E-07
Sr-90+D	0.00E+00	1.13E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95+D	2.72E-03	3.73E-03	3.01E-03	2.39E-03	2.24E-03	2.66E-03	2.43E-03
Ru-103+D	3.62E-05	5.32E-05	3.92E-05	3.15E-05	2.93E-05	3.69E-05	3.22E-05
Ru-106+D	3.16E-02	2.35E+00	3.44E-02	2.77E-02	2.59E-02	3.17E-02	2.83E-02
Sb-125+D	8.15E-03	1.42E-02	8.79E-03	6.88E-03	6.36E-03	8.23E-03	6.88E-03
I-129 (ELE)	9.19E-04	2.21E-03	9.07E-04	4.47E-04	3.52E-04	7.81E-04	1.68E-04
I-129 (ORG)	1.84E-05	4.41E-05	1.81E-05	8.94E-06	7.04E-06	1.56E-05	3.36E-06
I-131	2.85E-03	7.48E-03	3.09E-03	2.45E-03	2.27E-03	3.02E-03	2.52E-03
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	6.93E-02	1.21E-01	7.63E-02	6.10E-02	5.70E-02	6.84E-02	6.18E-02
Cs-137+D	6.70E-01	2.13E+00	7.37E-01	5.88E-01	5.48E-01	6.65E-01	5.99E-01
Cs-138	2.84E-06	2.11E-05	3.09E-06	2.52E-06	2.39E-06	2.71E-06	2.55E-06
Ce-144+D	4.59E-03	1.17E+00	5.05E-03	3.86E-03	3.59E-03	4.97E-03	3.64E-03
Pu-238	2.40E-06	2.01E-05	4.23E-07	3.42E-07	7.36E-08	5.74E-07	1.26E-07
Pu-239/240	1.87E-07	1.39E-06	5.75E-08	4.43E-08	2.40E-08	7.29E-08	2.77E-08
TOTALS	7.90E-01	1.71E+01	8.69E-01	6.93E-01	6.46E-01	7.85E-01	7.05E-01

Table C-26. (continued).

Report Date: 15-Apr-91
Infant receptor

1971 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	7.52E-03	7.48E-03	7.48E-03	7.48E-03	7.48E-03	7.48E-03	7.48E-03
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	4.70E-06	0.00E+00	0.00E+00	3.43E-06	8.68E-06	0.00E+00	3.61E-06
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	7.43E-40	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	4.80E-01	4.06E-02	4.06E-02	4.06E-02	4.06E-02	3.69E+00	2.62E+00
Zr-95	6.24E-05	4.11E-06	2.38E-06	4.11E-06	6.86E-06	1.44E-04	3.93E-05
Ru-103	2.54E-06	3.23E-07	3.62E-07	3.62E-07	4.94E-07	4.61E-07	5.27E-07
Ru-106	2.08E-02	5.14E-03	5.38E-03	5.38E-03	5.38E-03	6.31E-03	8.41E-03
Sb-125	9.69E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.22E-02	2.44E-05	4.11E-01	4.65E-05	2.33E-05	6.34E-05	3.78E-05
I-129 (ORG)	2.44E-04	4.89E-07	8.23E-03	9.31E-07	4.67E-07	1.27E-06	7.57E-07
I-131	1.59E-01	6.79E-04	5.35E+00	1.28E-03	6.63E-04	9.52E-04	1.01E-03
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	2.94E-02	2.00E-02	2.94E-02	2.58E-02	2.94E-02	3.05E-02	3.17E-02
Cs-137	7.44E-02	5.95E-02	7.07E-02	6.70E-02	7.07E-02	7.81E-02	1.00E-01
Cs-138	1.05E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	7.05E-03	1.41E-04	1.41E-04	1.41E-04	2.11E-03	6.07E-03	7.76E-03
Pu-238	4.48E-04	2.07E-05	2.07E-05	2.10E-05	2.24E-03	5.51E-03	7.57E-04
Pu-239/240	7.06E-05	3.02E-06	3.02E-06	3.02E-06	3.58E-04	9.07E-04	1.11E-04
TOTALS	7.91E-01	1.34E-01	5.92E+00	1.48E-01	1.59E-01	3.83E+00	2.78E+00

Table C-27. Dose evaluation for 1960 operational releases (adult)

Report Date: 15-Apr-91
FileName: IDA13A.WK1

Operational Releases

Calendar Year..... 1960
Receptor Location..... Atomic City
Receptor Designation..... Adult
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

5.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
19.1 : Distance to Receptor (km)
16.4 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
9.59E-01 : Breathing Rate (m**3/hr) 1.38E-87
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

6.08E+03

Report Date: 15-Apr-91

1960 Operational Releases

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	5.76E-01	1.40E+00	6.34E-01	4.97E-01	4.66E-01	5.84E-01	4.79E-01
Inhalation	1.21E-01	2.08E-06	2.13E-01	1.84E-01	3.60E-01	1.63E+00	1.35E-01
Grnd Surface	7.68E-02	2.35E+00	8.44E-02	6.73E-02	6.29E-02	7.63E-02	6.85E-02
Ingestion	2.42E-01	1.98E-05	5.15E+00	1.37E-02	9.58E-03	8.92E-01	3.90E-01
TOTALS	1.02E+00	3.75E+00	6.08E+00	7.61E-01	8.99E-01	3.18E+00	1.07E+00

Table C-27. (continued).

Report Date: 15-Apr-91
Adult receptor

1960 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	6.05E+00	3.94E-15	3.45E-11	0.00E+00
Na-24	1.50E+01	1.11E+01	6.84E-15	6.00E-11	1.80E-03
Ar-41	1.83E+00	1.04E+05	4.36E-11	3.82E-07	0.00E+00
Cr-51	6.65E+02	3.86E-02	2.51E-17	2.20E-13	1.80E-03
Co-60	4.62E+04	2.93E-01	1.91E-16	1.67E-12	1.80E-03
Kr-85	9.40E+04	6.13E+02	3.99E-13	3.50E-09	0.00E+00
Kr-85m	4.48E+00	4.72E+03	2.57E-12	2.25E-08	0.00E+00
Kr-87	1.27E+00	1.70E+04	5.86E-12	5.14E-08	0.00E+00
Kr-88	2.84E+00	1.67E+04	8.18E-12	7.17E-08	0.00E+00
Rb-89	2.57E-01	1.36E+03	3.84E-14	3.37E-10	1.80E-03
Sr-89	1.21E+03	9.38E-01	6.10E-16	5.35E-12	1.80E-03
Sr-90	2.51E+05	2.60E+00	1.69E-15	1.48E-11	1.80E-03
Sr-91	9.50E+00	3.85E-02	2.30E-17	2.02E-13	1.80E-03
Y-91	1.40E+03	1.02E+00	6.63E-16	5.81E-12	1.80E-03
Y-91m	8.29E-01	1.52E+00	3.73E-16	3.27E-12	1.80E-03
Zr-95	1.54E+03	1.03E+00	6.70E-16	5.87E-12	1.80E-03
Tc-99m	6.02E+00	3.39E+00	1.93E-15	1.69E-11	1.80E-03
Ru-103	9.44E+02	7.49E-01	4.87E-16	4.27E-12	1.80E-03
Ru-106	8.84E+03	8.11E-02	5.28E-17	4.63E-13	1.80E-03
Te-132	7.82E+01	1.77E+00	1.14E-15	9.99E-12	1.80E-03
I-129 (ELE)	1.38E+11	9.02E-09	5.87E-24	5.14E-20	1.80E-02
I-129 (ORG)	1.38E+11	1.80E-08	1.17E-23	1.03E-19	1.80E-04
I-131	1.93E+02	3.26E+01	2.11E-14	1.85E-10	1.80E-02
I-132	2.30E+00	1.76E+02	8.06E-14	7.07E-10	1.80E-02
I-133	2.08E+01	1.19E+01	7.42E-15	6.50E-11	1.80E-02
I-135	6.61E+00	1.35E-01	7.77E-17	6.82E-13	1.80E-02
Xe-131m	2.84E+02	2.34E+03	1.52E-12	1.33E-08	0.00E+00
Xe-133	1.26E+02	7.94E+05	5.13E-10	4.50E-06	0.00E+00
Xe-133m	5.26E+01	1.74E+04	1.11E-11	9.77E-08	0.00E+00
Xe-135	9.11E+00	3.98E+04	2.37E-11	2.08E-07	0.00E+00
Xe-135m	2.56E-01	9.97E+03	2.77E-13	2.43E-09	0.00E+00
Xe-138	2.36E-01	5.59E+04	1.18E-12	1.04E-08	0.00E+00
Cs-134	1.81E+04	2.16E-01	1.41E-16	1.23E-12	1.80E-03
Cs-136	3.16E+02	7.06E-02	4.58E-17	4.02E-13	1.80E-03
Cs-137	2.65E+05	9.58E-01	6.23E-16	5.46E-12	1.80E-03
Cs-138	5.37E-01	1.23E+03	1.78E-13	1.56E-09	1.80E-03
Ba-139	1.39E+00	1.16E+02	4.21E-14	3.69E-10	1.80E-03
Ba-140	3.07E+02	2.95E+00	1.91E-15	1.68E-11	1.80E-03
Ce-141	7.80E+02	1.63E+00	1.06E-15	9.30E-12	1.80E-03
Ce-144	6.82E+03	8.21E-01	5.34E-16	4.68E-12	1.80E-03
Pr-143	3.25E+02	2.87E+00	1.86E-15	1.63E-11	1.80E-03
Pm-147	2.30E+04	1.72E+00	1.12E-15	9.81E-12	1.80E-03
Hg-203	1.12E+03	4.36E+00	2.83E-15	2.49E-11	1.80E-03
Pu-238	7.69E+05	3.12E-02	2.03E-17	1.78E-13	1.80E-03
Pu-239	2.12E+08	2.85E-04	1.85E-19	1.63E-15	1.80E-03
Pu-239/240	2.12E+08	4.55E-03	2.96E-18	2.60E-14	1.80E-03
TOTAL		1.07E+06			

Table C-27. (continued).

Report Date: 15-Apr-91
Adult receptor

1960 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	2.19E-05	2.19E-05	2.19E-05	2.19E-05	2.19E-05	0.00E+00	2.19E-05
Na-24	4.63E-04	9.09E-04	4.27E-04	6.15E-04	3.41E-04	3.81E-04	4.16E-04
Ar-41	2.89E-01	5.31E-01	3.20E-01	2.57E-01	2.43E-01	2.74E-01	2.59E-01
Cr-51	5.66E-07	4.99E-07	4.05E-07	7.40E-07	2.96E-07	4.07E-07	3.75E-07
Co-60	1.46E-02	1.76E-02	1.45E-02	1.46E-02	1.33E-02	1.24E-02	1.27E-02
Kr-85	4.46E-06	6.30E-04	4.82E-06	3.89E-06	3.62E-06	4.54E-06	3.97E-06
Kr-85m	2.09E-03	6.72E-03	2.32E-03	1.73E-03	1.58E-03	2.54E-03	1.66E-03
Kr-87	2.62E-02	9.67E-02	2.80E-02	2.32E-02	2.20E-02	2.51E-02	2.37E-02
Kr-88	1.25E-01	3.42E-01	1.34E-01	1.12E-01	1.06E-01	1.18E-01	1.14E-01
Rb-89	4.62E-04	9.72E-04	4.90E-04	4.80E-04	3.78E-04	4.26E-04	4.07E-04
Sr-89	3.36E-04	2.85E-03	7.36E-08	1.59E-03	5.50E-08	6.38E-08	5.94E-08
Sr-90	8.76E-02	2.09E+00	0.00E+00	1.56E-01	0.00E+00	8.50E-01	3.72E-01
Sr-91	5.45E-07	1.43E-06	2.95E-07	1.74E-06	2.20E-07	2.62E-07	2.39E-07
Y-91	2.50E-04	3.74E-03	2.26E-06	2.06E-03	1.70E-06	1.93E-06	1.82E-06
Y-91m	1.21E-06	1.62E-06	1.21E-06	1.79E-06	9.03E-07	1.12E-06	9.90E-07
Zr-95	1.82E-03	2.27E-03	1.84E-03	1.46E-03	1.36E-03	3.71E-03	1.75E-03
Tc-99m	2.92E-06	3.00E-06	5.61E-06	3.32E-06	1.76E-06	3.04E-06	1.99E-06
Ru-103	2.07E-04	2.23E-04	1.64E-04	3.73E-04	1.23E-04	1.55E-04	1.35E-04
Ru-106	2.95E-04	4.76E-03	6.99E-05	1.70E-03	5.25E-05	6.44E-05	5.74E-05
Te-132	2.67E-04	4.99E-04	3.18E-03	2.06E-04	1.37E-04	1.66E-04	1.48E-04
I-129 (ELE)	4.10E-09	6.22E-10	1.28E-07	1.26E-10	9.93E-11	2.20E-10	4.73E-11
I-129 (ORG)	9.94E-11	1.24E-11	3.13E-09	2.51E-12	1.98E-12	4.40E-12	9.45E-13
I-131	1.67E-01	2.83E-02	5.33E+00	9.28E-03	8.61E-03	1.14E-02	9.55E-03
I-132	3.96E-03	1.08E-02	8.39E-03	3.96E-03	3.08E-03	3.65E-03	3.34E-03
I-133	1.56E-03	4.46E-03	3.15E-02	5.62E-04	5.25E-04	6.46E-04	5.74E-04
I-135	6.29E-06	1.77E-05	2.81E-05	5.95E-06	4.61E-06	5.25E-06	4.95E-06
Xe-131m	6.51E-05	8.51E-04	6.69E-05	3.96E-05	3.35E-05	6.42E-05	2.69E-05
Xe-133	9.03E-02	2.98E-01	1.03E-01	6.67E-02	5.95E-02	1.18E-01	4.35E-02
Xe-133m	1.66E-03	1.35E-02	1.77E-03	1.28E-03	1.16E-03	1.80E-03	1.20E-03
Xe-135	2.96E-02	8.62E-02	3.22E-02	2.49E-02	2.30E-02	3.32E-02	2.53E-02
Xe-135m	5.96E-04	9.89E-04	6.45E-04	5.18E-04	4.82E-04	6.04E-04	5.29E-04
Xe-138	7.40E-03	1.52E-02	7.97E-03	6.56E-03	6.22E-03	7.14E-03	6.66E-03
Cs-134	4.76E-03	4.59E-03	4.82E-03	4.24E-03	2.16E-03	4.41E-03	4.36E-03
Cs-136	3.34E-05	2.85E-05	3.44E-05	3.00E-05	1.71E-05	3.18E-05	3.09E-05
Cs-137	5.06E-02	1.36E-01	5.46E-02	4.51E-02	3.50E-02	5.00E-02	4.58E-02
Cs-138	2.52E-03	5.73E-03	2.62E-03	3.01E-03	2.02E-03	2.28E-03	2.17E-03
Ba-139	6.57E-05	6.68E-04	1.00E-05	3.37E-04	6.88E-06	1.07E-05	7.18E-06
Ba-140	1.26E-03	4.35E-03	1.14E-03	1.03E-03	8.76E-04	1.16E-03	1.01E-03
Ce-141	1.64E-04	1.50E-04	5.35E-05	6.00E-04	3.48E-05	6.01E-05	3.45E-05
Ce-144	1.94E-03	3.09E-02	1.34E-04	1.31E-02	9.51E-05	1.32E-04	9.65E-05
Pr-143	1.93E-04	9.04E-04	4.06E-12	7.52E-04	3.03E-12	3.61E-12	3.29E-12
Pm-147	3.61E-04	1.91E-07	7.80E-08	2.63E-03	5.02E-08	9.04E-08	4.80E-08
Hg-203	8.59E-03	6.73E-04	5.68E-04	4.02E-03	4.05E-04	5.86E-04	5.52E-03
Pu-238	8.01E-02	2.41E-05	5.07E-07	4.10E-07	3.13E-01	1.41E+00	1.17E-01
Pu-239	8.11E-04	1.04E-07	4.31E-09	3.32E-09	3.18E-03	1.48E-02	1.18E-03
Pu-239/240	1.30E-02	1.66E-06	6.89E-08	5.31E-08	5.07E-02	2.36E-01	1.88E-02

Table C-27. (continued).

Report Date: 15-Apr-91
Adult receptor

1960 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	1.63E-04	2.39E-04	1.74E-04	1.46E-04	1.39E-04	1.52E-04	1.48E-04
Ar-41	2.89E-01	5.31E-01	3.20E-01	2.57E-01	2.43E-01	2.74E-01	2.59E-01
Cr-51	3.96E-09	5.14E-09	4.29E-09	3.38E-09	3.13E-09	4.31E-09	3.48E-09
Co-60	2.48E-06	3.37E-06	2.74E-06	2.19E-06	2.08E-06	2.34E-06	2.21E-06
Kr-85	4.46E-06	6.30E-04	4.82E-06	3.89E-06	3.62E-06	4.54E-06	3.97E-06
Kr-85m	2.09E-03	6.72E-03	2.32E-03	1.73E-03	1.58E-03	2.54E-03	1.66E-03
Kr-87	2.62E-02	9.67E-02	2.80E-02	2.32E-02	2.20E-02	2.51E-02	2.37E-02
Kr-88+D	1.25E-01	3.42E-01	1.34E-01	1.12E-01	1.06E-01	1.18E-01	1.14E-01
Rb-89	4.38E-04	8.87E-04	4.76E-04	3.88E-04	3.68E-04	4.15E-04	3.96E-04
Sr-89	4.37E-10	2.67E-06	4.88E-10	3.86E-10	3.63E-10	4.23E-10	3.92E-10
Sr-90+D	0.00E+00	1.44E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.21E-07	2.80E-07	1.34E-07	1.07E-07	9.98E-08	1.19E-07	1.08E-07
Y-91	1.24E-08	3.03E-06	1.38E-08	1.10E-08	1.04E-08	1.18E-08	1.11E-08
Y-91m	1.00E-06	1.41E-06	1.09E-06	8.73E-07	8.13E-07	1.01E-06	8.92E-07
Zr-95+D	8.37E-06	1.17E-05	9.29E-06	7.37E-06	6.91E-06	8.21E-06	7.50E-06
Tc-99m	1.28E-06	1.62E-06	1.43E-06	1.04E-06	9.41E-07	1.63E-06	9.47E-07
Ru-103+D	1.18E-06	1.65E-06	1.28E-06	1.03E-06	9.54E-07	1.20E-06	1.05E-06
Ru-106+D	5.54E-08	6.75E-07	6.06E-08	4.86E-08	4.54E-08	5.54E-08	4.96E-08
Te-132+D	1.52E-05	2.47E-05	1.67E-05	1.33E-05	1.24E-05	1.50E-05	1.35E-05
I-129 (ELE)	2.45E-16	6.80E-16	2.43E-16	1.21E-16	9.68E-17	2.12E-16	4.59E-17
I-129 (ORG)	4.88E-16	1.36E-15	4.86E-16	2.41E-16	1.93E-16	4.24E-16	9.15E-17
I-131	4.03E-05	7.42E-05	4.37E-05	3.46E-05	3.21E-05	4.26E-05	3.56E-05
I-132	9.59E-04	1.57E-03	1.06E-03	8.46E-04	7.90E-04	9.35E-04	8.54E-04
I-133	2.28E-05	5.17E-05	2.48E-05	1.99E-05	1.86E-05	2.28E-05	2.03E-05
I-135	6.48E-07	1.06E-06	7.12E-07	5.76E-07	5.44E-07	6.18E-07	5.82E-07
Xe-131m	6.51E-05	8.51E-04	6.69E-05	3.96E-05	3.35E-05	6.42E-05	2.69E-05
Xe-133	9.03E-02	2.98E-01	1.03E-01	6.67E-02	5.95E-02	1.18E-01	4.35E-02
Xe-133m	1.66E-03	1.35E-02	1.77E-03	1.28E-03	1.16E-03	1.80E-03	1.20E-03
Xe-135	2.96E-02	8.62E-02	3.22E-02	2.49E-02	2.30E-02	3.32E-02	2.53E-02
Xe-135m	5.96E-04	9.89E-04	6.45E-04	5.18E-04	4.82E-04	6.04E-04	5.29E-04
Xe-138	7.40E-03	1.52E-02	7.97E-03	6.56E-03	6.22E-03	7.14E-03	6.66E-03
Cs-134	1.12E-06	1.64E-06	1.24E-06	9.89E-07	9.24E-07	1.11E-06	1.01E-06
Cs-136	5.17E-07	7.14E-07	5.72E-07	4.54E-07	4.25E-07	5.08E-07	4.58E-07
Cs-137+D	1.80E-06	3.26E-06	1.99E-06	1.58E-06	1.48E-06	1.79E-06	1.61E-06
Cs-138	2.26E-03	4.73E-03	2.47E-03	2.01E-03	1.90E-03	2.15E-03	2.04E-03
Ba-139	7.58E-06	3.03E-04	8.38E-06	6.28E-06	5.77E-06	8.97E-06	6.02E-06
Ba-140+D	2.89E-05	5.02E-05	3.14E-05	2.55E-05	2.41E-05	2.78E-05	2.60E-05
Ce-141	4.09E-07	1.26E-06	4.57E-07	3.30E-07	2.99E-07	5.14E-07	2.99E-07
Ce-144+D	1.43E-07	5.38E-06	1.56E-07	1.22E-07	1.14E-07	1.51E-07	1.18E-07
Pr-143	8.60E-14	3.85E-06	9.51E-14	7.56E-14	7.07E-14	8.43E-14	7.69E-14
Pm-147	2.05E-11	1.07E-07	2.30E-11	1.64E-11	1.49E-11	2.67E-11	1.42E-11
Hg-203	3.26E-06	5.10E-06	3.54E-06	2.74E-06	2.53E-06	3.65E-06	2.77E-06
Pu-238	8.95E-12	6.21E-11	3.47E-12	2.39E-12	1.45E-12	4.24E-12	1.08E-12
Pu-239	7.62E-14	2.69E-13	6.28E-14	4.48E-14	3.80E-14	7.43E-14	3.61E-14
Pu-239/240	1.22E-12	4.29E-12	1.00E-12	7.16E-13	6.07E-13	1.19E-12	5.77E-13
TOTALS	5.76E-01	1.40E+00	6.34E-01	4.97E-01	4.66E-01	5.84E-01	4.79E-01

Table C-27. (continued).

Report Date: 15-Apr-91
Adult receptor

1960 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	2.08E-06	2.08E-06	2.08E-06	2.08E-06	2.08E-06	0.00E+00	2.08E-06
Na-24	5.47E-05	0.00E+00	0.00E+00	2.53E-04	0.00E+00	0.00E+00	4.49E-05
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	5.48E-08	0.00E+00	0.00E+00	4.22E-07	0.00E+00	0.00E+00	0.00E+00
Co-60	2.40E-04	0.00E+00	0.00E+00	2.08E-03	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.20E-05	0.00E+00	0.00E+00	8.07E-05	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.90E-04	0.00E+00	0.00E+00	1.59E-03	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.85E-02	0.00E+00	0.00E+00	1.56E-01	0.00E+00	0.00E+00	0.00E+00
Sr-91	2.71E-07	0.00E+00	0.00E+00	1.51E-06	0.00E+00	0.00E+00	0.00E+00
Y-91	2.45E-04	0.00E+00	0.00E+00	2.06E-03	0.00E+00	0.00E+00	0.00E+00
Y-91m	9.73E-08	0.00E+00	0.00E+00	8.16E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.07E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.08E-03	2.70E-04
Tc-99m	5.19E-07	0.00E+00	2.92E-06	1.38E-06	0.00E+00	0.00E+00	2.11E-07
Ru-103	3.19E-05	0.00E+00	0.00E+00	2.41E-04	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.95E-04	0.00E+00	0.00E+00	1.64E-03	0.00E+00	0.00E+00	0.00E+00
Te-132	7.38E-05	0.00E+00	2.20E-03	6.04E-05	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	8.88E-12	0.00E+00	2.91E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	1.77E-11	0.00E+00	5.81E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	5.68E-03	0.00E+00	1.95E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	2.24E-04	0.00E+00	4.27E-03	6.78E-04	0.00E+00	0.00E+00	0.00E+00
I-133	3.37E-04	0.00E+00	1.12E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	7.19E-07	0.00E+00	2.03E-05	1.05E-06	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	5.55E-05	0.00E+00	4.84E-05	5.20E-05	0.00E+00	4.84E-05	5.20E-05
Cs-136	2.89E-06	0.00E+00	2.43E-06	3.27E-06	0.00E+00	2.43E-06	2.70E-06
Cs-137	1.68E-04	0.00E+00	1.52E-04	1.73E-04	0.00E+00	1.52E-04	1.62E-04
Cs-138	1.32E-04	0.00E+00	0.00E+00	8.82E-04	0.00E+00	0.00E+00	0.00E+00
Ba-139	5.67E-05	0.00E+00	0.00E+00	3.29E-04	0.00E+00	0.00E+00	0.00E+00
Ba-140	5.79E-05	0.00E+00	0.00E+00	1.01E-04	0.00E+00	1.43E-04	7.72E-05
Ce-141	7.58E-05	0.00E+00	0.00E+00	5.62E-04	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.57E-03	0.00E+00	0.00E+00	1.30E-02	0.00E+00	0.00E+00	0.00E+00
Pr-143	1.14E-04	0.00E+00	0.00E+00	7.52E-04	0.00E+00	0.00E+00	0.00E+00
Pm-147	3.20E-04	0.00E+00	0.00E+00	2.63E-03	0.00E+00	0.00E+00	0.00E+00
Hg-203	1.55E-04	0.00E+00	0.00E+00	9.77E-05	0.00E+00	0.00E+00	9.77E-05
Pu-238	7.85E-02	0.00E+00	0.00E+00	0.00E+00	3.07E-01	1.38E+00	1.14E-01
Pu-239	7.95E-04	0.00E+00	0.00E+00	0.00E+00	3.12E-03	1.45E-02	1.15E-03
Pu-239/240	1.27E-02	0.00E+00	0.00E+00	0.00E+00	4.98E-02	2.31E-01	1.84E-02
TOTALS	1.21E-01	2.08E-06	2.13E-01	1.84E-01	3.60E-01	1.63E+00	1.35E-01

Table C-27. (continued).

Report Date: 15-Apr-91
Adult receptor

1960 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.38E-04	6.70E-04	2.53E-04	2.12E-04	2.02E-04	2.22E-04	2.16E-04
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	3.72E-07	4.94E-07	4.00E-07	3.15E-07	2.92E-07	4.03E-07	3.26E-07
Co-60	1.31E-02	1.76E-02	1.45E-02	1.16E-02	1.10E-02	1.24E-02	1.17E-02
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.23E-05	8.54E-05	1.35E-05	1.09E-05	1.03E-05	1.17E-05	1.11E-05
Sr-89	6.58E-08	2.85E-03	7.31E-08	5.81E-08	5.47E-08	6.34E-08	5.90E-08
Sr-90+D	0.00E+00	2.09E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.46E-07	1.15E-06	1.61E-07	1.28E-07	1.20E-07	1.43E-07	1.30E-07
Y-91	2.02E-06	3.74E-03	2.24E-06	1.79E-06	1.69E-06	1.92E-06	1.81E-06
Y-91m	1.10E-07	2.08E-07	1.20E-07	9.65E-08	9.00E-08	1.11E-07	9.85E-08
Zr-95+D	1.65E-03	2.26E-03	1.83E-03	1.45E-03	1.36E-03	1.61E-03	1.47E-03
Tc-99m	1.12E-06	1.38E-06	1.24E-06	9.03E-07	8.20E-07	1.42E-06	8.27E-07
Ru-103+D	1.51E-04	2.21E-04	1.63E-04	1.31E-04	1.22E-04	1.54E-04	1.34E-04
Ru-106+D	6.40E-05	4.76E-03	6.98E-05	5.61E-05	5.25E-05	6.44E-05	5.73E-05
Te-132+D	1.52E-04	4.75E-04	1.68E-04	1.32E-04	1.24E-04	1.51E-04	1.35E-04
I-129 (ELE)	2.59E-10	6.22E-10	2.56E-10	1.26E-10	9.93E-11	2.20E-10	4.73E-11
I-129 (ORG)	5.17E-12	1.24E-11	5.10E-12	2.51E-12	1.98E-12	4.39E-12	9.45E-13
I-131	1.08E-02	2.82E-02	1.17E-02	9.24E-03	8.58E-03	1.14E-02	9.51E-03
I-132	2.78E-03	9.19E-03	3.07E-03	2.44E-03	2.29E-03	2.72E-03	2.49E-03
I-133	6.19E-04	4.40E-03	6.76E-04	5.42E-04	5.07E-04	6.24E-04	5.54E-04
I-135	4.87E-06	1.66E-05	5.34E-06	4.33E-06	4.06E-06	4.63E-06	4.37E-06
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	2.63E-03	4.59E-03	2.89E-03	2.31E-03	2.16E-03	2.59E-03	2.34E-03
Cs-136	2.03E-05	2.78E-05	2.25E-05	1.78E-05	1.67E-05	2.01E-05	1.80E-05
Cs-137+D	4.28E-02	1.36E-01	4.71E-02	3.75E-02	3.50E-02	4.24E-02	3.82E-02
Cs-138	1.34E-04	9.99E-04	1.46E-04	1.19E-04	1.13E-04	1.28E-04	1.21E-04
Ba-139	1.48E-06	3.65E-04	1.64E-06	1.22E-06	1.11E-06	1.76E-06	1.16E-06
Ba-140+D	1.02E-03	4.30E-03	1.11E-03	9.01E-04	8.52E-04	9.85E-04	9.10E-04
Ce-141	4.75E-05	1.49E-04	5.31E-05	3.80E-05	3.45E-05	5.96E-05	3.42E-05
Ce-144+D	1.22E-04	3.09E-02	1.34E-04	1.02E-04	9.50E-05	1.32E-04	9.64E-05
Pr-143	3.59E-12	9.00E-04	3.97E-12	3.16E-12	2.96E-12	3.53E-12	3.22E-12
Pm-147	6.90E-08	8.35E-08	7.79E-08	5.56E-08	5.02E-08	9.04E-08	4.80E-08
Hg-203	5.18E-04	6.68E-04	5.64E-04	4.36E-04	4.02E-04	5.83E-04	4.40E-04
Pu-238	2.88E-06	2.41E-05	5.07E-07	4.10E-07	8.84E-08	6.89E-07	1.52E-07
Pu-239	1.41E-08	1.04E-07	4.31E-09	3.32E-09	1.80E-09	5.47E-09	2.07E-09
Pu-239/240	2.24E-07	1.66E-06	6.89E-08	5.31E-08	2.88E-08	8.73E-08	3.31E-08
TOTALS	7.68E-02	2.35E+00	8.44E-02	6.73E-02	6.29E-02	7.63E-02	6.85E-02

Table C-27. (continued).

Report Date: 15-Apr-91
Adult receptor

1960 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.98E-05	1.98E-05	1.98E-05	1.98E-05	1.98E-05	0.00E+00	1.98E-05
Na-24	6.04E-06	0.00E+00	0.00E+00	4.14E-06	0.00E+00	7.34E-06	6.04E-06
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.35E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.56E-08
Co-60	1.26E-03	0.00E+00	0.00E+00	9.22E-04	2.33E-03	0.00E+00	9.71E-04
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.04E-36	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.47E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	6.91E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.50E-01	3.72E-01
Sr-91	6.85E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.33E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.38E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	5.37E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	2.84E-09	0.00E+00	1.47E-08	0.00E+00	0.00E+00	0.00E+00	1.09E-09
Ru-103	2.31E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	3.55E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	2.65E-05	0.00E+00	7.88E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	3.83E-09	0.00E+00	1.27E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	7.65E-11	0.00E+00	2.54E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	1.51E-01	0.00E+00	5.12E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	4.56E-08	0.00E+00	1.12E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	5.77E-04	0.00E+00	1.96E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	5.30E-08	0.00E+00	1.77E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	2.08E-03	0.00E+00	1.88E-03	1.88E-03	0.00E+00	1.77E-03	1.97E-03
Cs-136	9.71E-06	0.00E+00	8.83E-06	8.47E-06	0.00E+00	8.83E-06	9.71E-06
Cs-137	7.68E-03	0.00E+00	7.37E-03	7.37E-03	0.00E+00	7.37E-03	7.37E-03
Cs-138	3.17E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	8.27E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.55E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	3.99E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	2.45E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	7.91E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	4.06E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Hg-203	7.91E-03	0.00E+00	0.00E+00	3.48E-03	0.00E+00	0.00E+00	4.98E-03
Pu-238	1.57E-03	0.00E+00	0.00E+00	0.00E+00	6.20E-03	2.77E-02	2.31E-03
Pu-239	1.62E-05	0.00E+00	0.00E+00	0.00E+00	6.05E-05	2.95E-04	2.23E-05
Pu-239/240	2.59E-04	0.00E+00	0.00E+00	0.00E+00	9.65E-04	4.71E-03	3.56E-04
TOTALS	2.42E-01	1.98E-05	5.15E+00	1.37E-02	9.58E-03	8.92E-01	3.90E-01

Table C-28. Dose evaluation for 1960 operational releases (infant)

Report Date: 15-Apr-91
FileName: IDA13B.WK1

Operational Releases

Calendar Year..... 1960
Receptor Location..... Atomic City
Receptor Designation..... Infant
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

5.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
19.1 : Distance to Receptor (km)
16.4 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

FileName: IDA13B.WK1

1960 Operational Releases

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surfac	Red Marrow
Immersion	5.76E-01	1.40E+00	6.34E-01	4.97E-01	4.66E-01	5.84E-01	4.79E-01
Inhalation	3.37E-02	1.09E-03	2.59E-01	4.73E-02	1.00E-01	2.52E-01	3.79E-02
Grnd Surface	7.68E-02	2.35E+00	8.44E-02	6.73E-02	6.29E-02	7.63E-02	6.85E-02
Ingestion	6.97E-01	1.47E-02	2.02E+01	1.84E-02	1.94E-02	7.04E-01	5.01E-01
TOTALS	1.38E+00	3.76E+00	2.12E+01	6.30E-01	6.49E-01	1.62E+00	1.09E+00

Table C-28. (continued).

Report Date: 15-Apr-91
Infant receptor

1960 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	6.05E+00	3.94E-15	3.45E-11	0.00E+00
Na-24	1.50E+01	1.11E+01	6.84E-15	6.00E-11	1.80E-03
Ar-41	1.83E+00	1.04E+05	4.36E-11	3.82E-07	0.00E+00
Cr-51	6.65E+02	3.86E-02	2.51E-17	2.20E-13	1.80E-03
Co-60	4.62E+04	2.93E-01	1.91E-16	1.67E-12	1.80E-03
Kr-85	9.40E+04	6.13E+02	3.99E-13	3.50E-09	0.00E+00
Kr-85m	4.48E+00	4.72E+03	2.57E-12	2.25E-08	0.00E+00
Kr-87	1.27E+00	1.70E+04	5.86E-12	5.14E-08	0.00E+00
Kr-88	2.84E+00	1.67E+04	8.18E-12	7.17E-08	0.00E+00
Rb-89	2.57E-01	1.36E+03	3.84E-14	3.37E-10	1.80E-03
Sr-89	1.21E+03	9.38E-01	6.10E-16	5.35E-12	1.80E-03
Sr-90	2.51E+05	2.60E+00	1.69E-15	1.48E-11	1.80E-03
Sr-91	9.50E+00	3.85E-02	2.30E-17	2.02E-13	1.80E-03
Y-91	1.40E+03	1.02E+00	6.63E-16	5.81E-12	1.80E-03
Y-91m	8.29E-01	1.52E+00	3.73E-16	3.27E-12	1.80E-03
Zr-95	1.54E+03	1.03E+00	6.70E-16	5.87E-12	1.80E-03
Tc-99m	6.02E+00	3.39E+00	1.93E-15	1.69E-11	1.80E-03
Ru-103	9.44E+02	7.49E-01	4.87E-16	4.27E-12	1.80E-03
Ru-106	8.84E+03	8.11E-02	5.28E-17	4.63E-13	1.80E-03
Te-132	7.82E+01	1.77E+00	1.14E-15	9.99E-12	1.80E-03
I-129 (ELE)	1.38E+11	9.02E-09	5.87E-24	5.14E-20	1.80E-02
I-129 (ORG)	1.38E+11	1.80E-08	1.17E-23	1.03E-19	1.80E-04
I-131	1.93E+02	3.26E+01	2.11E-14	1.85E-10	1.80E-02
I-132	2.30E+00	1.76E+02	8.06E-14	7.07E-10	1.80E-02
I-133	2.08E+01	1.19E+01	7.42E-15	6.50E-11	1.80E-02
I-135	6.61E+00	1.35E-01	7.77E-17	6.82E-13	1.80E-02
Xe-131m	2.84E+02	2.34E+03	1.52E-12	1.33E-08	0.00E+00
Xe-133	1.26E+02	7.94E+05	5.13E-10	4.50E-06	0.00E+00
Xe-133m	5.26E+01	1.74E+04	1.11E-11	9.77E-08	0.00E+00
Xe-135	9.11E+00	3.98E+04	2.37E-11	2.08E-07	0.00E+00
Xe-135m	2.56E-01	9.97E+03	2.77E-13	2.43E-09	0.00E+00
Xe-138	2.36E-01	5.59E+04	1.18E-12	1.04E-08	0.00E+00
Cs-134	1.81E+04	2.16E-01	1.41E-16	1.23E-12	1.80E-03
Cs-136	3.16E+02	7.06E-02	4.58E-17	4.02E-13	1.80E-03
Cs-137	2.65E+05	9.58E-01	6.23E-16	5.46E-12	1.80E-03
Cs-138	5.37E-01	1.23E+03	1.78E-13	1.56E-09	1.80E-03
Ba-139	1.39E+00	1.16E+02	4.21E-14	3.69E-10	1.80E-03
Ba-140	3.07E+02	2.95E+00	1.91E-15	1.68E-11	1.80E-03
Ce-141	7.80E+02	1.63E+00	1.06E-15	9.30E-12	1.80E-03
Ce-144	6.82E+03	8.21E-01	5.34E-16	4.68E-12	1.80E-03
Pr-143	3.25E+02	2.87E+00	1.86E-15	1.63E-11	1.80E-03
Pm-147	2.30E+04	1.72E+00	1.12E-15	9.81E-12	1.80E-03
Hg-203	1.12E+03	4.36E+00	2.83E-15	2.49E-11	1.80E-03
Pu-238	7.69E+05	3.12E-02	2.03E-17	1.78E-13	1.80E-03
Pu-239	2.12E+08	2.85E-04	1.85E-19	1.63E-15	1.80E-03
Pu-239/240	2.12E+08	4.55E-03	2.96E-18	2.60E-14	1.80E-03
TOTAL		1.07E+06			

Table C-28. (continued).

Report Date: 15-Apr-91
Infant receptor

1960 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.49E-05	1.48E-05	1.48E-05	1.48E-05	1.48E-05	1.48E-05	1.48E-05
Na-24	4.15E-04	9.09E-04	4.27E-04	4.02E-04	3.41E-04	3.78E-04	3.76E-04
Ar-41	2.89E-01	5.31E-01	3.20E-01	2.57E-01	2.43E-01	2.74E-01	2.59E-01
Cr-51	4.09E-07	4.99E-07	4.05E-07	3.88E-07	2.96E-07	4.07E-07	3.38E-07
Co-60	1.34E-02	1.76E-02	1.45E-02	1.22E-02	1.15E-02	1.24E-02	1.19E-02
Kr-85	4.46E-06	6.30E-04	4.82E-06	3.89E-06	3.62E-06	4.54E-06	3.97E-06
Kr-85m	2.09E-03	6.72E-03	2.32E-03	1.73E-03	1.58E-03	2.54E-03	1.66E-03
Kr-87	2.62E-02	9.67E-02	2.80E-02	2.32E-02	2.20E-02	2.51E-02	2.37E-02
Kr-88	1.25E-01	3.42E-01	1.34E-01	1.12E-01	1.06E-01	1.18E-01	1.14E-01
Rb-89	4.52E-04	9.72E-04	4.90E-04	4.12E-04	3.78E-04	4.26E-04	4.07E-04
Sr-89	6.85E-05	2.85E-03	7.36E-08	2.62E-04	5.50E-08	6.38E-08	5.94E-08
Sr-90	9.02E-02	2.10E+00	7.63E-03	7.74E-03	7.63E-03	6.94E-01	4.93E-01
Sr-91	3.12E-07	1.43E-06	2.95E-07	4.84E-07	2.20E-07	2.62E-07	2.39E-07
Y-91	5.07E-05	3.74E-03	2.26E-06	3.42E-04	1.70E-06	1.93E-06	1.82E-06
Y-91m	1.13E-06	1.62E-06	1.21E-06	1.10E-06	9.03E-07	1.12E-06	9.90E-07
Zr-95	1.86E-03	2.29E-03	1.86E-03	2.56E-03	1.41E-03	1.80E-03	1.54E-03
Tc-99m	2.49E-06	3.00E-06	3.16E-06	2.17E-06	1.76E-06	3.04E-06	1.81E-06
Ru-103	2.17E-04	2.28E-04	1.71E-04	4.83E-04	1.34E-04	1.63E-04	1.42E-04
Ru-106	3.72E-04	4.77E-03	8.54E-05	2.18E-03	6.88E-05	8.26E-05	8.09E-05
Te-132	1.80E-04	4.99E-04	5.75E-04	1.56E-04	1.37E-04	1.66E-04	1.48E-04
I-129 (ELE)	3.71E-09	6.29E-10	1.16E-07	1.39E-10	1.06E-10	2.38E-10	5.80E-11
I-129 (ORG)	7.81E-11	1.26E-11	2.46E-09	2.93E-12	2.12E-12	4.77E-12	1.17E-12
I-131	6.18E-01	3.09E-02	2.04E+01	1.49E-02	1.12E-02	1.51E-02	1.34E-02
I-132	4.15E-03	1.08E-02	1.16E-02	4.44E-03	3.14E-03	3.70E-03	3.39E-03
I-133	9.02E-04	4.46E-03	9.49E-03	5.62E-04	5.25E-04	6.46E-04	5.74E-04
I-135	5.64E-06	1.77E-05	9.56E-06	5.08E-06	4.61E-06	5.25E-06	4.95E-06
Xe-131m	6.51E-05	8.51E-04	6.69E-05	3.96E-05	3.35E-05	6.42E-05	2.69E-05
Xe-133	9.03E-02	2.98E-01	1.03E-01	6.67E-02	5.95E-02	1.18E-01	4.35E-02
Xe-133m	1.66E-03	1.35E-02	1.77E-03	1.28E-03	1.16E-03	1.80E-03	1.20E-03
Xe-135	2.96E-02	8.62E-02	3.22E-02	2.49E-02	2.30E-02	3.32E-02	2.53E-02
Xe-135m	5.96E-04	9.89E-04	6.45E-04	5.18E-04	4.82E-04	6.04E-04	5.29E-04
Xe-138	7.40E-03	1.52E-02	7.97E-03	6.56E-03	6.22E-03	7.14E-03	6.66E-03
Cs-134	3.75E-03	5.35E-03	4.02E-03	3.31E-03	3.29E-03	3.76E-03	3.56E-03
Cs-136	2.50E-05	2.85E-05	2.70E-05	2.21E-05	1.71E-05	2.44E-05	2.27E-05
Cs-137	4.76E-02	1.40E-01	5.16E-02	4.19E-02	3.96E-02	4.75E-02	4.47E-02
Cs-138	2.41E-03	5.73E-03	2.62E-03	2.27E-03	2.02E-03	2.28E-03	2.17E-03
Ba-139	1.84E-05	6.68E-04	1.00E-05	6.18E-05	6.88E-06	1.07E-05	7.18E-06
Ba-140	1.06E-03	4.35E-03	1.14E-03	9.43E-04	8.76E-04	1.04E-03	9.49E-04
Ce-141	6.16E-05	1.50E-04	5.35E-05	1.31E-04	3.48E-05	6.01E-05	3.45E-05
Ce-144	2.47E-03	3.09E-02	1.46E-04	1.70E-02	3.02E-04	5.55E-04	6.03E-04
Pr-143	2.04E-05	9.04E-04	4.06E-12	1.24E-04	3.03E-12	3.61E-12	3.29E-12
Pm-147	5.63E-05	1.91E-07	7.80E-08	4.34E-04	5.02E-08	9.04E-08	4.80E-08
Hg-203	1.71E-03	6.73E-04	5.68E-04	9.64E-04	4.05E-04	5.86E-04	1.19E-03
Pu-238	1.93E-02	8.08E-04	7.84E-04	1.98E-02	8.79E-02	2.15E-01	2.80E-02
Pu-239	1.86E-04	7.16E-06	7.06E-06	1.71E-04	8.83E-04	2.16E-03	2.65E-04
Pu-240	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-28. (continued).

Report Date: 15-Apr-91
Infant receptor

1960 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	1.63E-04	2.39E-04	1.74E-04	1.46E-04	1.39E-04	1.52E-04	1.48E-04
Ar-41	2.89E-01	5.31E-01	3.20E-01	2.57E-01	2.43E-01	2.74E-01	2.59E-01
Cr-51	3.96E-09	5.14E-09	4.29E-09	3.38E-09	3.13E-09	4.31E-09	3.48E-09
Co-60	2.48E-06	3.37E-06	2.74E-06	2.19E-06	2.08E-06	2.34E-06	2.21E-06
Kr-85	4.46E-06	6.30E-04	4.82E-06	3.89E-06	3.62E-06	4.54E-06	3.97E-06
Kr-85m	2.09E-03	6.72E-03	2.32E-03	1.73E-03	1.58E-03	2.54E-03	1.66E-03
Kr-87	2.62E-02	9.67E-02	2.80E-02	2.32E-02	2.20E-02	2.51E-02	2.37E-02
Kr-88+D	1.25E-01	3.42E-01	1.34E-01	1.12E-01	1.06E-01	1.18E-01	1.14E-01
Rb-89	4.38E-04	8.87E-04	4.76E-04	3.88E-04	3.68E-04	4.15E-04	3.96E-04
Sr-89	4.37E-10	2.67E-06	4.88E-10	3.86E-10	3.63E-10	4.23E-10	3.92E-10
Sr-90+D	0.00E+00	1.44E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.21E-07	2.80E-07	1.34E-07	1.07E-07	9.98E-08	1.19E-07	1.08E-07
Y-91	1.24E-08	3.03E-06	1.38E-08	1.10E-08	1.04E-08	1.18E-08	1.11E-08
Y-91m	1.00E-06	1.41E-06	1.09E-06	8.73E-07	8.13E-07	1.01E-06	8.92E-07
Zr-95+D	8.37E-06	1.17E-05	9.29E-06	7.37E-06	6.91E-06	8.21E-06	7.50E-06
Tc-99m	1.28E-06	1.62E-06	1.43E-06	1.04E-06	9.41E-07	1.63E-06	9.47E-07
Ru-103+D	1.18E-06	1.65E-06	1.28E-06	1.03E-06	9.54E-07	1.20E-06	1.05E-06
Ru-106+D	5.54E-08	6.75E-07	6.06E-08	4.86E-08	4.54E-08	5.54E-08	4.96E-08
Te-132+D	1.52E-05	2.47E-05	1.67E-05	1.33E-05	1.24E-05	1.50E-05	1.35E-05
I-129 (ELE)	2.45E-16	6.80E-16	2.43E-16	1.21E-16	9.68E-17	2.12E-16	4.59E-17
I-129 (ORG)	4.88E-16	1.36E-15	4.86E-16	2.41E-16	1.93E-16	4.24E-16	9.15E-17
I-131	4.03E-05	7.42E-05	4.37E-05	3.46E-05	3.21E-05	4.26E-05	3.56E-05
I-132	9.59E-04	1.57E-03	1.06E-03	8.46E-04	7.90E-04	9.35E-04	8.54E-04
I-133	2.28E-05	5.17E-05	2.48E-05	1.99E-05	1.86E-05	2.28E-05	2.03E-05
I-135	6.48E-07	1.06E-06	7.12E-07	5.76E-07	5.44E-07	6.18E-07	5.82E-07
Xe-131m	6.51E-05	8.51E-04	6.69E-05	3.96E-05	3.35E-05	6.42E-05	2.69E-05
Xe-133	9.03E-02	2.98E-01	1.03E-01	6.67E-02	5.95E-02	1.18E-01	4.35E-02
Xe-133m	1.66E-03	1.35E-02	1.77E-03	1.28E-03	1.16E-03	1.80E-03	1.20E-03
Xe-135	2.96E-02	8.62E-02	3.22E-02	2.49E-02	2.30E-02	3.32E-02	2.53E-02
Xe-135m	5.96E-04	9.89E-04	6.45E-04	5.18E-04	4.82E-04	6.04E-04	5.29E-04
Xe-138	7.40E-03	1.52E-02	7.97E-03	6.56E-03	6.22E-03	7.14E-03	6.66E-03
Cs-134	1.12E-06	1.64E-06	1.24E-06	9.89E-07	9.24E-07	1.11E-06	1.01E-06
Cs-136	5.17E-07	7.14E-07	5.72E-07	4.54E-07	4.25E-07	5.08E-07	4.58E-07
Cs-137+D	1.80E-06	3.26E-06	1.99E-06	1.58E-06	1.48E-06	1.79E-06	1.61E-06
Cs-138	2.26E-03	4.73E-03	2.47E-03	2.01E-03	1.90E-03	2.15E-03	2.04E-03
Ba-139	7.58E-06	3.03E-04	8.38E-06	6.28E-06	5.77E-06	8.97E-06	6.02E-06
Ba-140+D	2.89E-05	5.02E-05	3.14E-05	2.55E-05	2.41E-05	2.78E-05	2.60E-05
Ce-141	4.09E-07	1.26E-06	4.57E-07	3.30E-07	2.99E-07	5.14E-07	2.99E-07
Ce-144+D	1.43E-07	5.38E-06	1.56E-07	1.22E-07	1.14E-07	1.51E-07	1.18E-07
Pr-143	8.60E-14	3.85E-06	9.51E-14	7.56E-14	7.07E-14	8.43E-14	7.69E-14
Pm-147	2.05E-11	1.07E-07	2.30E-11	1.64E-11	1.49E-11	2.67E-11	1.42E-11
Hg-203	3.26E-06	5.10E-06	3.54E-06	2.74E-06	2.53E-06	3.65E-06	2.77E-06
Pu-238	8.95E-12	6.21E-11	3.47E-12	2.39E-12	1.45E-12	4.24E-12	1.08E-12
Pu-239	7.62E-14	2.69E-13	6.28E-14	4.48E-14	3.80E-14	7.43E-14	3.61E-14
Pu-240	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	5.76E-01	1.40E+00	6.34E-01	4.97E-01	4.66E-01	5.84E-01	4.79E-01

Table C-28. (continued).

Report Date: 15-Apr-91
Infant receptor

1960 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.11E-06	1.11E-06	1.11E-06	1.11E-06	1.11E-06	1.11E-06	1.11E-06
Na-24	9.01E-06	0.00E+00	0.00E+00	4.17E-05	0.00E+00	0.00E+00	7.39E-06
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	9.03E-09	0.00E+00	0.00E+00	6.95E-08	0.00E+00	0.00E+00	0.00E+00
Co-60	3.96E-05	0.00E+00	0.00E+00	3.43E-04	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.97E-06	0.00E+00	0.00E+00	1.33E-05	0.00E+00	0.00E+00	0.00E+00
Sr-89	3.13E-05	0.00E+00	0.00E+00	2.62E-04	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.04E-03	8.67E-05	8.67E-05	1.99E-04	8.67E-05	8.25E-03	5.91E-03
Sr-91	4.46E-08	0.00E+00	0.00E+00	2.49E-07	0.00E+00	0.00E+00	0.00E+00
Y-91	4.04E-05	0.00E+00	0.00E+00	3.40E-04	0.00E+00	0.00E+00	0.00E+00
Y-91m	1.60E-08	0.00E+00	0.00E+00	1.34E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.72E-04	1.31E-05	2.58E-05	1.09E-03	4.46E-05	8.58E-05	3.37E-05
Tc-99m	8.55E-08	0.00E+00	4.81E-07	2.27E-07	0.00E+00	0.00E+00	3.47E-08
Ru-103	5.49E-05	3.24E-06	5.49E-06	3.49E-04	8.77E-06	5.99E-06	4.99E-06
Ru-106	2.65E-04	4.06E-06	4.60E-06	2.11E-03	5.41E-06	5.41E-06	6.43E-06
Te-132	1.22E-05	0.00E+00	3.63E-04	9.95E-06	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	2.11E-12	4.21E-15	6.92E-11	7.82E-14	4.21E-15	1.15E-14	6.62E-15
I-129 (ORG)	4.20E-12	8.40E-15	1.38E-10	1.56E-13	8.40E-15	2.29E-14	1.32E-14
I-131	7.45E-03	3.24E-05	2.49E-01	7.57E-04	3.57E-05	4.76E-05	4.88E-05
I-132	4.13E-04	3.26E-05	7.44E-03	1.16E-03	5.78E-05	4.96E-05	4.96E-05
I-133	5.55E-05	0.00E+00	1.85E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	1.18E-07	0.00E+00	3.34E-06	1.72E-07	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.22E-05	7.92E-06	1.15E-05	1.51E-05	1.15E-05	1.15E-05	1.22E-05
Cs-136	4.76E-07	0.00E+00	4.00E-07	5.39E-07	0.00E+00	4.00E-07	4.44E-07
Cs-137	4.15E-05	3.16E-05	3.83E-05	6.39E-05	3.83E-05	4.15E-05	5.43E-05
Cs-138	2.17E-05	0.00E+00	0.00E+00	1.45E-04	0.00E+00	0.00E+00	0.00E+00
Ba-139	9.34E-06	0.00E+00	0.00E+00	5.43E-05	0.00E+00	0.00E+00	0.00E+00
Ba-140	9.55E-06	0.00E+00	0.00E+00	1.67E-05	0.00E+00	2.36E-05	1.27E-05
Ce-141	1.25E-05	0.00E+00	0.00E+00	9.26E-05	0.00E+00	0.00E+00	0.00E+00
Ce-144	2.16E-03	7.12E-06	8.51E-06	1.69E-02	1.51E-04	2.63E-04	3.01E-04
Pr-143	1.88E-05	0.00E+00	0.00E+00	1.24E-04	0.00E+00	0.00E+00	0.00E+00
Pm-147	5.27E-05	0.00E+00	0.00E+00	4.34E-04	0.00E+00	0.00E+00	0.00E+00
Hg-203	2.55E-05	0.00E+00	0.00E+00	1.61E-05	0.00E+00	0.00E+00	1.61E-05
Pu-238	1.87E-02	7.59E-04	7.59E-04	1.98E-02	8.52E-02	2.08E-01	2.70E-02
Pu-239	1.81E-04	6.83E-06	6.83E-06	1.71E-04	8.56E-04	2.09E-03	2.57E-04
Pu-239/240	2.88E-03	1.09E-04	1.09E-04	2.73E-03	1.37E-02	3.34E-02	4.10E-03
TOTALS	3.37E-02	1.09E-03	2.59E-01	4.73E-02	1.00E-01	2.52E-01	3.79E-02

Table C-28. (continued).

Report Date: 15-Apr-91
Infant receptor

1960 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.38E-04	6.70E-04	2.53E-04	2.12E-04	2.02E-04	2.22E-04	2.16E-04
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	3.72E-07	4.94E-07	4.00E-07	3.15E-07	2.92E-07	4.03E-07	3.26E-07
Co-60	1.31E-02	1.76E-02	1.45E-02	1.16E-02	1.10E-02	1.24E-02	1.17E-02
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.23E-05	8.54E-05	1.35E-05	1.09E-05	1.03E-05	1.17E-05	1.11E-05
Sr-89	6.58E-08	2.85E-03	7.31E-08	5.81E-08	5.47E-08	6.34E-08	5.90E-08
Sr-90+D	0.00E+00	2.09E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.46E-07	1.15E-06	1.61E-07	1.28E-07	1.20E-07	1.43E-07	1.30E-07
Y-91	2.02E-06	3.74E-03	2.24E-06	1.79E-06	1.69E-06	1.92E-06	1.81E-06
Y-91m	1.10E-07	2.08E-07	1.20E-07	9.65E-08	9.00E-08	1.11E-07	9.85E-08
Zr-95+D	1.65E-03	2.26E-03	1.83E-03	1.45E-03	1.36E-03	1.61E-03	1.47E-03
Tc-99m	1.12E-06	1.38E-06	1.24E-06	9.03E-07	8.20E-07	1.42E-06	8.27E-07
Ru-103+D	1.51E-04	2.21E-04	1.63E-04	1.31E-04	1.22E-04	1.54E-04	1.34E-04
Ru-106+D	6.40E-05	4.76E-03	6.98E-05	5.61E-05	5.25E-05	6.44E-05	5.73E-05
Te-132+D	1.52E-04	4.75E-04	1.68E-04	1.32E-04	1.24E-04	1.51E-04	1.35E-04
I-129 (ELE)	2.59E-10	6.22E-10	2.56E-10	1.26E-10	9.93E-11	2.20E-10	4.73E-11
I-129 (ORG)	5.17E-12	1.24E-11	5.10E-12	2.51E-12	1.98E-12	4.39E-12	9.45E-13
I-131	1.08E-02	2.82E-02	1.17E-02	9.24E-03	8.58E-03	1.14E-02	9.51E-03
I-132	2.78E-03	9.19E-03	3.07E-03	2.44E-03	2.29E-03	2.72E-03	2.49E-03
I-133	6.19E-04	4.40E-03	6.76E-04	5.42E-04	5.07E-04	6.24E-04	5.54E-04
I-135	4.87E-06	1.66E-05	5.34E-06	4.33E-06	4.06E-06	4.63E-06	4.37E-06
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	2.63E-03	4.59E-03	2.89E-03	2.31E-03	2.16E-03	2.59E-03	2.34E-03
Cs-136	2.03E-05	2.78E-05	2.25E-05	1.78E-05	1.67E-05	2.01E-05	1.80E-05
Cs-137+D	4.28E-02	1.36E-01	4.71E-02	3.75E-02	3.50E-02	4.24E-02	3.82E-02
Cs-138	1.34E-04	9.99E-04	1.46E-04	1.19E-04	1.13E-04	1.28E-04	1.21E-04
Ba-139	1.48E-06	3.65E-04	1.64E-06	1.22E-06	1.11E-06	1.76E-06	1.16E-06
Ba-140+D	1.02E-03	4.30E-03	1.11E-03	9.01E-04	8.52E-04	9.85E-04	9.10E-04
Ce-141	4.75E-05	1.49E-04	5.31E-05	3.80E-05	3.45E-05	5.96E-05	3.42E-05
Ce-144+D	1.22E-04	3.09E-02	1.34E-04	1.02E-04	9.50E-05	1.32E-04	9.64E-05
Pr-143	3.59E-12	9.00E-04	3.97E-12	3.16E-12	2.96E-12	3.53E-12	3.22E-12
Pm-147	6.90E-08	8.35E-08	7.79E-08	5.56E-08	5.02E-08	9.04E-08	4.80E-08
Hg-203	5.18E-04	6.68E-04	5.64E-04	4.36E-04	4.02E-04	5.83E-04	4.40E-04
Pu-238	2.88E-06	2.41E-05	5.07E-07	4.10E-07	8.84E-08	6.89E-07	1.52E-07
Pu-239	1.41E-08	1.04E-07	4.31E-09	3.32E-09	1.80E-09	5.47E-09	2.07E-09
Pu-239/240	2.24E-07	1.66E-06	6.89E-08	5.31E-08	2.88E-08	8.73E-08	3.31E-08
TOTALS	7.68E-02	2.35E+00	8.44E-02	6.73E-02	6.29E-02	7.63E-02	6.85E-02

Table C-28. (continued).

Report Date: 15-Apr-91
Infant receptor

1960 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.38E-05	1.37E-05	1.37E-05	1.37E-05	1.37E-05	1.37E-05	1.37E-05
Na-24	3.58E-06	0.00E+00	0.00E+00	2.45E-06	0.00E+00	4.35E-06	3.58E-06
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	2.37E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.01E-09
Co-60	2.70E-04	0.00E+00	0.00E+00	1.97E-04	4.98E-04	0.00E+00	2.08E-04
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	6.31E-38	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	3.72E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	8.92E-02	7.54E-03	7.54E-03	7.54E-03	7.54E-03	6.86E-01	4.88E-01
Sr-91	3.01E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	8.22E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	2.15E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	3.78E-05	2.49E-06	1.44E-06	2.49E-06	4.16E-06	8.70E-05	2.38E-05
Tc-99m	2.35E-10	0.00E+00	1.22E-09	0.00E+00	0.00E+00	0.00E+00	9.02E-11
Ru-103	1.05E-05	1.34E-06	1.51E-06	1.51E-06	2.05E-06	1.92E-06	2.19E-06
Ru-106	4.22E-05	1.04E-05	1.09E-05	1.09E-05	1.09E-05	1.28E-05	1.70E-05
Te-132	9.11E-07	0.00E+00	2.71E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	3.45E-09	6.89E-12	1.16E-07	1.31E-11	6.58E-12	1.79E-11	1.07E-11
I-129 (ORG)	6.87E-11	1.37E-13	2.31E-09	2.62E-13	1.31E-13	3.56E-13	2.13E-13
I-131	6.00E-01	2.56E-03	2.02E+01	4.85E-03	2.50E-03	3.59E-03	3.82E-03
I-132	8.75E-09	7.29E-10	2.00E-07	1.46E-09	1.17E-09	1.02E-09	1.17E-09
I-133	2.04E-04	0.00E+00	6.94E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	5.20E-09	0.00E+00	1.74E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.11E-03	7.56E-04	1.11E-03	9.79E-04	1.11E-03	1.16E-03	1.20E-03
Cs-136	3.80E-06	0.00E+00	3.45E-06	3.31E-06	0.00E+00	3.45E-06	3.80E-06
Cs-137	4.75E-03	3.80E-03	4.51E-03	4.28E-03	4.51E-03	4.99E-03	6.41E-03
Cs-138	4.96E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	1.29E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	8.48E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.14E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.87E-04	3.73E-06	3.73E-06	3.73E-06	5.60E-05	1.61E-04	2.06E-04
Pr-143	1.54E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	3.57E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Hg-203	1.16E-03	0.00E+00	0.00E+00	5.09E-04	0.00E+00	0.00E+00	7.29E-04
Pu-238	5.37E-04	2.48E-05	2.48E-05	2.52E-05	2.69E-03	6.61E-03	9.09E-04
Pu-239	5.29E-06	2.27E-07	2.27E-07	2.27E-07	2.69E-05	6.80E-05	8.32E-06
Pu-239/240	8.45E-05	3.62E-06	3.62E-06	3.62E-06	4.29E-04	1.09E-03	1.33E-04
TOTALS	6.97E-01	1.47E-02	2.02E+01	1.84E-02	1.94E-02	7.04E-01	5.01E-01

Table C-29. Dose evaluation for 1957 operational releases (adult)

Report Date: 15-Apr-91
FileName: IDA13A.WK1

Operational Releases

Calendar Year..... 1957
Receptor Location..... Atomic City
Receptor Designation..... Adult
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

5.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
19.1 : Distance to Receptor (km)
16.4 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
9.59E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

1.38E-87

6.08E+03

Report Date: 15-Apr-91

1957 Operational Releases

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	2.42E-01	6.96E-01	2.68E-01	2.07E-01	1.93E-01	2.52E-01	1.95E-01
Inhalation	9.14E-01	1.90E-03	8.96E+00	3.47E+00	8.95E-01	4.09E+00	3.48E-01
Grnd Surface	2.24E+00	3.81E+01	2.46E+00	1.96E+00	1.83E+00	2.26E+00	2.00E+00
Ingestion	7.80E+00	1.80E-02	2.21E+02	3.01E-01	3.67E-02	1.20E+01	5.41E+00
TOTALS	1.12E+01	3.88E+01	2.33E+02	5.94E+00	2.95E+00	1.86E+01	7.95E+00

Table C-29. (continued).

Report Date: 15-Apr-91
Adult receptor

1957 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	5.51E+03	3.59E-12	3.14E-08	0.00E+00
Na-24	1.50E+01	3.66E+00	2.26E-15	1.98E-11	1.80E-03
Ar-41	1.83E+00	3.42E+04	1.43E-11	1.25E-07	0.00E+00
Cr-51	6.65E+02	1.27E-02	8.25E-18	7.24E-14	1.80E-03
Co-60	4.62E+04	9.62E-02	6.26E-17	5.49E-13	1.80E-03
Kr-85	9.40E+04	1.00E+05	6.52E-11	5.71E-07	0.00E+00
Kr-85m	4.48E+00	1.56E+03	8.46E-13	7.42E-09	0.00E+00
Kr-87	1.27E+00	5.59E+03	1.93E-12	1.69E-08	0.00E+00
Kr-88	2.84E+00	5.48E+03	2.68E-12	2.35E-08	0.00E+00
Rb-89	2.57E-01	4.48E+02	1.27E-14	1.11E-10	1.80E-03
Sr-89	1.21E+03	1.92E+01	1.25E-14	1.09E-10	1.80E-03
Sr-90	2.51E+05	3.57E+01	2.32E-14	2.04E-10	1.80E-03
Sr-91	9.50E+00	9.19E-01	5.49E-16	4.81E-12	1.80E-03
Y-91	1.40E+03	2.11E+01	1.37E-14	1.20E-10	1.80E-03
Y-91m	8.29E-01	5.01E-01	1.23E-16	1.08E-12	1.80E-03
Zr-95	1.54E+03	2.13E+01	1.39E-14	1.21E-10	1.80E-03
Tc-99m	6.02E+00	1.11E+00	6.32E-16	5.54E-12	1.80E-03
Ru-103	9.44E+02	1.53E+01	9.95E-15	8.72E-11	1.80E-03
Ru-106	8.84E+03	5.87E+00	3.82E-15	3.35E-11	1.80E-03
Te-132	7.82E+01	3.41E+01	2.20E-14	1.93E-10	1.80E-03
I-129 (ELE)	1.38E+11	4.30E-07	2.80E-22	2.45E-18	1.80E-02
I-129 (ORG)	1.38E+11	8.60E-07	5.60E-22	4.91E-18	1.80E-04
I-131	1.93E+02	1.40E+03	9.07E-13	7.95E-09	1.80E-02
I-132	2.30E+00	4.03E+03	1.85E-12	1.62E-08	1.80E-02
I-133	2.08E+01	4.41E+02	2.76E-13	2.42E-09	1.80E-02
I-135	6.61E+00	6.65E+00	3.83E-15	3.36E-11	1.80E-02
Xe-131m	2.84E+02	1.65E+03	1.07E-12	9.38E-09	0.00E+00
Xe-133	1.26E+02	4.63E+05	2.99E-10	2.62E-06	0.00E+00
Xe-133m	5.26E+01	9.16E+03	5.87E-12	5.15E-08	0.00E+00
Xe-135	9.11E+00	1.83E+04	1.09E-11	9.57E-08	0.00E+00
Xe-135m	2.56E-01	3.28E+03	9.12E-14	8.00E-10	0.00E+00
Xe-138	2.36E-01	1.83E+04	3.87E-13	3.39E-09	0.00E+00
Cs-134	1.81E+04	6.07E-01	3.95E-16	3.46E-12	1.80E-03
Cs-136	3.16E+02	3.17E+00	2.06E-15	1.80E-11	1.80E-03
Cs-137	2.65E+05	3.59E+01	2.34E-14	2.05E-10	1.80E-03
Cs-138	5.37E-01	4.04E+02	5.84E-14	5.12E-10	1.80E-03
Ba-139	1.39E+00	3.82E+01	1.39E-14	1.22E-10	1.80E-03
Ba-140	3.07E+02	5.85E+01	3.80E-14	3.33E-10	1.80E-03
Ce-141	7.80E+02	3.32E+01	2.16E-14	1.89E-10	1.80E-03
Ce-144	6.82E+03	5.40E+01	3.51E-14	3.08E-10	1.80E-03
Pr-143	3.25E+02	5.79E+01	3.76E-14	3.29E-10	1.80E-03
Pm-147	2.30E+04	1.40E+02	9.11E-14	7.99E-10	1.80E-03
Eu-154	7.71E+04	7.05E-02	4.59E-17	4.02E-13	1.80E-03
Hg-203	1.12E+03	1.43E+00	9.30E-16	8.15E-12	1.80E-03
Pu-238	7.69E+05	7.78E-02	5.06E-17	4.44E-13	1.80E-03
Pu-239	2.12E+08	9.36E-05	6.09E-20	5.34E-16	1.80E-03
Pu-239/240	2.12E+08	1.15E-02	7.48E-18	6.56E-14	1.80E-03
TOTAL		6.73E+05			

Table C-29. (continued).

Report Date: 15-Apr-91
Adult receptor

1957 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.99E-02	1.99E-02	1.99E-02	1.99E-02	1.99E-02	0.00E+00	1.99E-02
Na-24	1.53E-04	3.00E-04	1.41E-04	2.03E-04	1.12E-04	1.26E-04	1.37E-04
Ar-41	9.48E-02	1.74E-01	1.05E-01	8.42E-02	7.96E-02	8.99E-02	8.49E-02
Cr-51	1.86E-07	1.64E-07	1.33E-07	2.44E-07	9.72E-08	1.34E-07	1.23E-07
Co-60	4.78E-03	5.76E-03	4.76E-03	4.80E-03	4.37E-03	4.08E-03	4.17E-03
Kr-85	7.29E-04	1.03E-01	7.88E-04	6.35E-04	5.91E-04	7.42E-04	6.49E-04
Kr-85m	6.91E-04	2.22E-03	7.64E-04	5.69E-04	5.21E-04	8.37E-04	5.46E-04
Kr-87	8.61E-03	3.18E-02	9.19E-03	7.61E-03	7.23E-03	8.27E-03	7.79E-03
Kr-88	4.11E-02	1.12E-01	4.41E-02	3.67E-02	3.49E-02	3.88E-02	3.74E-02
Rb-89	1.52E-04	3.20E-04	1.61E-04	1.58E-04	1.25E-04	1.40E-04	1.34E-04
Sr-89	6.88E-03	5.84E-02	1.51E-06	3.25E-02	1.13E-06	1.31E-06	1.22E-06
Sr-90	1.20E+00	2.87E+01	0.00E+00	2.15E+00	0.00E+00	1.17E+01	5.10E+00
Sr-91	1.30E-05	3.42E-05	7.04E-06	4.16E-05	5.25E-06	6.27E-06	5.70E-06
Y-91	5.17E-03	7.74E-02	4.67E-05	4.27E-02	3.51E-05	3.99E-05	3.77E-05
Y-91m	3.98E-07	5.35E-07	3.99E-07	5.89E-07	2.98E-07	3.69E-07	3.26E-07
Zr-95	3.75E-02	4.70E-02	3.80E-02	3.02E-02	2.82E-02	7.66E-02	3.62E-02
Tc-99m	9.57E-07	9.82E-07	1.84E-06	1.09E-06	5.76E-07	9.97E-07	6.50E-07
Ru-103	4.23E-03	4.56E-03	3.35E-03	7.63E-03	2.51E-03	3.16E-03	2.76E-03
Ru-106	2.13E-02	3.44E-01	5.06E-03	1.23E-01	3.80E-03	4.66E-03	4.15E-03
Te-132	5.15E-03	9.62E-03	6.12E-02	3.97E-03	2.63E-03	3.20E-03	2.85E-03
I-129 (ELE)	1.95E-07	2.96E-08	6.09E-06	6.01E-09	4.73E-09	1.05E-08	2.26E-09
I-129 (ORG)	4.75E-09	5.93E-10	1.49E-07	1.20E-10	9.47E-11	2.10E-10	4.51E-11
I-131	7.19E+00	1.22E+00	2.29E+02	3.99E-01	3.70E-01	4.92E-01	4.10E-01
I-132	9.07E-02	2.46E-01	1.92E-01	9.07E-02	7.06E-02	8.36E-02	7.65E-02
I-133	5.79E-02	1.66E-01	1.17E+00	2.09E-02	1.96E-02	2.41E-02	2.14E-02
I-135	3.10E-04	8.71E-04	1.38E-03	2.93E-04	2.27E-04	2.59E-04	2.44E-04
Xe-131m	4.59E-05	6.00E-04	4.72E-05	2.79E-05	2.36E-05	4.53E-05	1.89E-05
Xe-133	5.26E-02	1.74E-01	6.01E-02	3.89E-02	3.47E-02	6.87E-02	2.54E-02
Xe-133m	8.74E-04	7.10E-03	9.33E-04	6.75E-04	6.10E-04	9.50E-04	6.33E-04
Xe-135	1.36E-02	3.97E-02	1.48E-02	1.15E-02	1.06E-02	1.53E-02	1.17E-02
Xe-135m	1.96E-04	3.25E-04	2.12E-04	1.70E-04	1.59E-04	1.99E-04	1.74E-04
Xe-138	2.42E-03	4.99E-03	2.61E-03	2.15E-03	2.04E-03	2.34E-03	2.18E-03
Cs-134	1.34E-02	1.29E-02	1.36E-02	1.19E-02	6.07E-03	1.24E-02	1.23E-02
Cs-136	1.50E-03	1.28E-03	1.54E-03	1.35E-03	7.67E-04	1.43E-03	1.39E-03
Cs-137	1.90E+00	5.10E+00	2.05E+00	1.69E+00	1.31E+00	1.87E+00	1.72E+00
Cs-138	8.29E-04	1.88E-03	8.60E-04	9.89E-04	6.62E-04	7.49E-04	7.11E-04
Ba-139	2.17E-05	2.20E-04	3.30E-06	1.11E-04	2.27E-06	3.53E-06	2.37E-06
Ba-140	2.50E-02	8.64E-02	2.27E-02	2.04E-02	1.74E-02	2.29E-02	2.01E-02
Ce-141	3.33E-03	3.05E-03	1.09E-03	1.22E-02	7.07E-04	1.22E-03	7.01E-04
Ce-144	1.27E-01	2.03E+00	8.80E-03	8.63E-01	6.26E-03	8.67E-03	6.35E-03
Pr-143	3.90E-03	1.82E-02	8.19E-11	1.52E-02	6.11E-11	7.29E-11	6.64E-11
Pm-147	2.93E-02	1.55E-05	6.35E-06	2.14E-01	4.08E-06	7.36E-06	3.91E-06
Eu-154	2.87E-03	6.75E-03	3.04E-03	2.52E-03	2.91E-03	3.47E-03	2.57E-03
Hg-203	2.82E-03	2.21E-04	1.86E-04	1.32E-03	1.33E-04	1.92E-04	1.81E-03
Pu-238	2.00E-01	6.01E-05	1.26E-06	1.02E-06	7.81E-01	3.52E+00	2.91E-01
Pu-239	2.66E-04	3.42E-08	1.42E-09	1.09E-09	1.04E-03	4.86E-03	3.86E-04
Pu-239/240	3.27E-02	4.20E-06	1.74E-07	1.34E-07	1.28E-01	5.97E-01	4.74E-02

Table C-29. (continued).

Report Date: 15-Apr-91
Adult receptor

1957 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	5.39E-05	7.89E-05	5.73E-05	4.80E-05	4.58E-05	5.01E-05	4.89E-05
Ar-41	9.48E-02	1.74E-01	1.05E-01	8.42E-02	7.96E-02	8.99E-02	8.49E-02
Cr-51	1.30E-09	1.69E-09	1.41E-09	1.11E-09	1.03E-09	1.42E-09	1.15E-09
Co-60	8.13E-07	1.11E-06	9.01E-07	7.19E-07	6.82E-07	7.69E-07	7.26E-07
Kr-85	7.29E-04	1.03E-01	7.88E-04	6.35E-04	5.91E-04	7.42E-04	6.49E-04
Kr-85m	6.91E-04	2.22E-03	7.64E-04	5.69E-04	5.21E-04	8.37E-04	5.46E-04
Kr-87	8.61E-03	3.18E-02	9.19E-03	7.61E-03	7.23E-03	8.27E-03	7.79E-03
Kr-88+D	4.11E-02	1.12E-01	4.41E-02	3.67E-02	3.49E-02	3.88E-02	3.74E-02
Rb-89	1.44E-04	2.92E-04	1.57E-04	1.28E-04	1.21E-04	1.37E-04	1.30E-04
Sr-89	8.95E-09	5.46E-05	9.99E-09	7.91E-09	7.42E-09	8.66E-09	8.02E-09
Sr-90+D	0.00E+00	1.97E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	2.90E-06	6.68E-06	3.20E-06	2.55E-06	2.38E-06	2.84E-06	2.59E-06
Y-91	2.56E-07	6.27E-05	2.85E-07	2.28E-07	2.15E-07	2.44E-07	2.30E-07
Y-91m	3.30E-07	4.66E-07	3.59E-07	2.88E-07	2.68E-07	3.32E-07	2.94E-07
Zr-95+D	1.73E-04	2.41E-04	1.92E-04	1.52E-04	1.43E-04	1.70E-04	1.55E-04
Tc-99m	4.18E-07	5.30E-07	4.68E-07	3.40E-07	3.08E-07	5.32E-07	3.10E-07
Ru-103+D	2.41E-05	3.37E-05	2.60E-05	2.10E-05	1.95E-05	2.46E-05	2.15E-05
Ru-106+D	4.01E-06	4.89E-05	4.39E-06	3.52E-06	3.29E-06	4.01E-06	3.59E-06
Te-132+D	2.93E-04	4.76E-04	3.22E-04	2.57E-04	2.40E-04	2.90E-04	2.59E-04
I-129 (ELE)	1.17E-14	3.24E-14	1.16E-14	5.76E-15	4.61E-15	1.01E-14	2.19E-15
I-129 (ORG)	2.33E-14	6.49E-14	2.32E-14	1.15E-14	9.23E-15	2.02E-14	4.37E-15
I-131	1.73E-03	3.19E-03	1.88E-03	1.49E-03	1.38E-03	1.83E-03	1.53E-03
I-132	2.20E-02	3.60E-02	2.42E-02	1.94E-02	1.81E-02	2.14E-02	1.96E-02
I-133	8.47E-04	1.92E-03	9.24E-04	7.42E-04	6.93E-04	8.50E-04	7.56E-04
I-135	3.19E-05	5.24E-05	3.51E-05	2.84E-05	2.68E-05	3.04E-05	2.87E-05
Xe-131m	4.59E-05	6.00E-04	4.72E-05	2.79E-05	2.36E-05	4.53E-05	1.89E-05
Xe-133	5.26E-02	1.74E-01	6.01E-02	3.89E-02	3.47E-02	6.87E-02	2.54E-02
Xe-133m	8.74E-04	7.10E-03	9.33E-04	6.75E-04	6.10E-04	9.50E-04	6.33E-04
Xe-135	1.36E-02	3.97E-02	1.48E-02	1.15E-02	1.06E-02	1.53E-02	1.17E-02
Xe-135m	1.96E-04	3.25E-04	2.12E-04	1.70E-04	1.59E-04	1.99E-04	1.74E-04
Xe-138	2.42E-03	4.99E-03	2.61E-03	2.15E-03	2.04E-03	2.34E-03	2.18E-03
Cs-134	3.16E-06	4.62E-06	3.49E-06	2.78E-06	2.60E-06	3.12E-06	2.83E-06
Cs-136	2.32E-05	3.21E-05	2.57E-05	2.04E-05	1.91E-05	2.28E-05	2.06E-05
Cs-137+D	6.76E-05	1.22E-04	7.44E-05	5.92E-05	5.54E-05	6.71E-05	6.05E-05
Cs-138	7.42E-04	1.55E-03	8.12E-04	6.60E-04	6.25E-04	7.07E-04	6.72E-04
Ba-139	2.50E-06	9.99E-05	2.76E-06	2.07E-06	1.90E-06	2.95E-06	1.98E-06
Ba-140+D	5.72E-04	9.96E-04	6.23E-04	5.07E-04	4.78E-04	5.52E-04	5.16E-04
Ce-141	8.32E-06	2.57E-05	9.30E-06	6.71E-06	6.08E-06	1.05E-05	6.08E-06
Ce-144+D	9.39E-06	3.54E-04	1.03E-05	8.02E-06	7.50E-06	9.96E-06	7.74E-06
Pr-143	1.74E-12	7.77E-05	1.92E-12	1.52E-12	1.43E-12	1.70E-12	1.55E-12
Pm-147	1.67E-09	8.73E-06	1.88E-09	1.34E-09	1.21E-09	2.18E-09	1.16E-09
Eu-154	2.98E-07	4.63E-07	3.31E-07	2.62E-07	2.46E-07	2.92E-07	2.64E-07
Hg-203	1.07E-06	1.67E-06	1.16E-06	8.98E-07	8.29E-07	1.20E-06	9.07E-07
Pu-238	2.23E-11	1.55E-10	8.65E-12	5.97E-12	3.62E-12	1.06E-11	2.69E-12
Pu-239	2.50E-14	8.83E-14	2.06E-14	1.47E-14	1.25E-14	2.44E-14	1.19E-14
Pu-239/240	3.07E-12	1.08E-11	2.53E-12	1.81E-12	1.53E-12	3.00E-12	1.46E-12
TOTALS	2.42E-01	6.96E-01	2.68E-01	2.07E-01	1.93E-01	2.52E-01	1.95E-01

Table C-29. (continued).

Report Date: 15-Apr-91
Adult receptor

1957 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.90E-03	1.90E-03	1.90E-03	1.90E-03	1.90E-03	0.00E+00	1.90E-03
Na-24	1.80E-05	0.00E+00	0.00E+00	8.35E-05	0.00E+00	0.00E+00	1.48E-05
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.80E-08	0.00E+00	0.00E+00	1.39E-07	0.00E+00	0.00E+00	0.00E+00
Co-60	7.89E-05	0.00E+00	0.00E+00	6.84E-04	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	3.94E-06	0.00E+00	0.00E+00	2.66E-05	0.00E+00	0.00E+00	0.00E+00
Sr-89	3.88E-03	0.00E+00	0.00E+00	3.25E-02	0.00E+00	0.00E+00	0.00E+00
Sr-90	2.54E-01	0.00E+00	0.00E+00	2.15E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91	6.46E-06	0.00E+00	0.00E+00	3.60E-05	0.00E+00	0.00E+00	0.00E+00
Y-91	5.07E-03	0.00E+00	0.00E+00	4.27E-02	0.00E+00	0.00E+00	0.00E+00
Y-91m	3.21E-08	0.00E+00	0.00E+00	2.69E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	2.21E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.31E-02	5.59E-03
Tc-99m	1.70E-07	0.00E+00	9.56E-07	4.51E-07	0.00E+00	0.00E+00	6.90E-08
Ru-103	6.52E-04	0.00E+00	0.00E+00	4.93E-03	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.41E-02	0.00E+00	0.00E+00	1.19E-01	0.00E+00	0.00E+00	0.00E+00
Te-132	1.42E-03	0.00E+00	4.25E-02	1.16E-03	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	4.23E-10	0.00E+00	1.39E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	8.47E-10	0.00E+00	2.78E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	2.44E-01	0.00E+00	8.39E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	5.12E-03	0.00E+00	9.78E-02	1.55E-02	0.00E+00	0.00E+00	0.00E+00
I-133	1.25E-02	0.00E+00	4.18E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	3.54E-05	0.00E+00	9.98E-04	5.15E-05	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.56E-04	0.00E+00	1.36E-04	1.46E-04	0.00E+00	1.36E-04	1.46E-04
Cs-136	1.30E-04	0.00E+00	1.09E-04	1.47E-04	0.00E+00	1.09E-04	1.21E-04
Cs-137	6.28E-03	0.00E+00	5.69E-03	6.48E-03	0.00E+00	5.69E-03	6.09E-03
Cs-138	4.32E-05	0.00E+00	0.00E+00	2.90E-04	0.00E+00	0.00E+00	0.00E+00
Ba-139	1.87E-05	0.00E+00	0.00E+00	1.08E-04	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.15E-03	0.00E+00	0.00E+00	2.01E-03	0.00E+00	2.84E-03	1.53E-03
Ce-141	1.54E-03	0.00E+00	0.00E+00	1.14E-02	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.03E-01	0.00E+00	0.00E+00	8.56E-01	0.00E+00	0.00E+00	0.00E+00
Pr-143	2.31E-03	0.00E+00	0.00E+00	1.52E-02	0.00E+00	0.00E+00	0.00E+00
Pm-147	2.60E-02	0.00E+00	0.00E+00	2.14E-01	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.00E-04	0.00E+00	0.00E+00	1.12E-04	6.17E-04	7.33E-04	1.58E-04
Hg-203	5.08E-05	0.00E+00	0.00E+00	3.20E-05	0.00E+00	0.00E+00	3.20E-05
Pu-238	1.96E-01	0.00E+00	0.00E+00	0.00E+00	7.66E-01	3.45E+00	2.85E-01
Pu-239	2.61E-04	0.00E+00	0.00E+00	0.00E+00	1.02E-03	4.76E-03	3.79E-04
Pu-239/240	3.21E-02	0.00E+00	0.00E+00	0.00E+00	1.26E-01	5.85E-01	4.65E-02
TOTALS	9.14E-01	1.90E-03	8.96E+00	3.47E+00	8.95E-01	4.09E+00	3.48E-01

Table C-29. (continued).

Report Date: 15-Apr-91
Adult receptor

1957 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	7.86E-05	2.21E-04	8.35E-05	6.98E-05	6.67E-05	7.31E-05	7.13E-05
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.22E-07	1.63E-07	1.32E-07	1.04E-07	9.62E-08	1.32E-07	1.07E-07
Co-60	4.29E-03	5.76E-03	4.76E-03	3.82E-03	3.61E-03	4.08E-03	3.85E-03
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.05E-06	2.81E-05	4.43E-06	3.60E-06	3.41E-06	3.86E-06	3.66E-06
Sr-89	1.35E-06	5.83E-02	1.50E-06	1.19E-06	1.12E-06	1.30E-06	1.21E-06
Sr-90+D	0.00E+00	2.87E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	3.48E-06	2.76E-05	3.84E-06	3.06E-06	2.87E-06	3.42E-06	3.11E-06
Y-91	4.17E-05	7.74E-02	4.64E-05	3.71E-05	3.49E-05	3.97E-05	3.74E-05
Y-91m	3.64E-08	6.87E-08	3.96E-08	3.18E-08	2.97E-08	3.67E-08	3.25E-08
Zr-95+D	3.40E-02	4.68E-02	3.78E-02	3.00E-02	2.81E-02	3.34E-02	3.05E-02
Tc-99m	3.68E-07	4.52E-07	4.07E-07	2.96E-07	2.68E-07	4.65E-07	2.71E-07
Ru-103+D	3.08E-03	4.52E-03	3.33E-03	2.67E-03	2.49E-03	3.14E-03	2.74E-03
Ru-106+D	4.64E-03	3.44E-01	5.06E-03	4.06E-03	3.80E-03	4.66E-03	4.15E-03
Te-132+D	2.93E-03	9.14E-03	3.23E-03	2.55E-03	2.39E-03	2.91E-03	2.59E-03
I-129 (ELE)	1.24E-08	2.96E-08	1.22E-08	6.01E-09	4.73E-09	1.05E-08	2.26E-09
I-129 (ORG)	2.47E-10	5.93E-10	2.44E-10	1.20E-10	9.47E-11	2.10E-10	4.51E-11
I-131	4.63E-01	1.21E+00	5.02E-01	3.97E-01	3.69E-01	4.90E-01	4.09E-01
I-132	6.36E-02	2.10E-01	7.02E-02	5.58E-02	5.25E-02	6.22E-02	5.69E-02
I-133	2.31E-02	1.64E-01	2.52E-02	2.02E-02	1.89E-02	2.32E-02	2.06E-02
I-135	2.40E-04	8.19E-04	2.63E-04	2.13E-04	2.00E-04	2.28E-04	2.15E-04
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	7.38E-03	1.29E-02	8.13E-03	6.49E-03	6.07E-03	7.29E-03	6.59E-03
Cs-136	9.10E-04	1.25E-03	1.01E-03	7.99E-04	7.48E-04	9.01E-04	8.08E-04
Cs-137+D	1.60E+00	5.10E+00	1.76E+00	1.41E+00	1.31E+00	1.59E+00	1.43E+00
Cs-138	4.41E-05	3.28E-04	4.80E-05	3.92E-05	3.71E-05	4.20E-05	3.96E-05
Ba-139	4.88E-07	1.20E-04	5.39E-07	4.01E-07	3.67E-07	5.78E-07	3.82E-07
Ba-140+D	2.02E-02	8.54E-02	2.21E-02	1.79E-02	1.69E-02	1.95E-02	1.81E-02
Ce-141	9.67E-04	3.03E-03	1.08E-03	7.73E-04	7.01E-04	1.21E-03	6.95E-04
Ce-144+D	7.99E-03	2.03E+00	8.79E-03	6.73E-03	6.25E-03	8.66E-03	6.34E-03
Pr-143	7.25E-11	1.82E-02	8.00E-11	6.38E-11	5.97E-11	7.12E-11	6.49E-11
Pm-147	5.62E-06	6.80E-06	6.34E-06	4.52E-06	4.08E-06	7.36E-06	3.91E-06
Eu-154	2.75E-03	6.75E-03	3.04E-03	2.41E-03	2.26E-03	2.70E-03	2.41E-03
Hg-203	1.70E-04	2.19E-04	1.85E-04	1.43E-04	1.32E-04	1.91E-04	1.44E-04
Pu-238	7.19E-06	6.01E-05	1.26E-06	1.02E-06	2.20E-07	1.72E-06	3.78E-07
Pu-239	4.62E-09	3.42E-08	1.42E-09	1.09E-09	5.92E-10	1.80E-09	6.81E-10
Pu-239/240	5.67E-07	4.20E-06	1.74E-07	1.34E-07	7.28E-08	2.21E-07	8.37E-08
TOTALS	2.24E+00	3.81E+01	2.46E+00	1.96E+00	1.83E+00	2.26E+00	2.00E+00

Table C-29. (continued).

Report Date: 15-Apr-91
Adult receptor

1957 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	0.00E+00	1.80E-02
Na-24	1.99E-06	0.00E+00	0.00E+00	1.37E-06	0.00E+00	2.42E-06	1.99E-06
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	4.44E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.50E-08
Co-60	4.14E-04	0.00E+00	0.00E+00	3.03E-04	7.65E-04	0.00E+00	3.19E-04
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.33E-36	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	3.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	9.48E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.17E+01	5.10E+00
Sr-91	1.63E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	4.81E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	4.54E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.11E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	9.31E-10	0.00E+00	4.81E-09	0.00E+00	0.00E+00	0.00E+00	3.57E-10
Ru-103	4.73E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	2.57E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	5.11E-04	0.00E+00	1.52E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.83E-07	0.00E+00	6.07E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ORG)	3.65E-09	0.00E+00	1.21E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	6.48E+00	0.00E+00	2.20E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	1.04E-06	0.00E+00	2.56E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	2.15E-02	0.00E+00	7.30E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.61E-06	0.00E+00	8.74E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	5.84E-03	0.00E+00	5.29E-03	5.29E-03	0.00E+00	4.97E-03	5.52E-03
Cs-136	4.36E-04	0.00E+00	3.96E-04	3.81E-04	0.00E+00	3.96E-04	4.36E-04
Cs-137	2.88E-01	0.00E+00	2.76E-01	2.76E-01	0.00E+00	2.76E-01	2.76E-01
Cs-138	1.04E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	2.72E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	3.08E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	8.12E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.61E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	1.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	3.30E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.70E-05	0.00E+00	0.00E+00	0.00E+00	2.62E-05	3.18E-05	7.66E-06
Hg-203	2.59E-03	0.00E+00	0.00E+00	1.14E-03	0.00E+00	0.00E+00	1.63E-03
Pu-238	3.92E-03	0.00E+00	0.00E+00	0.00E+00	1.55E-02	6.91E-02	5.77E-03
Pu-239	5.34E-06	0.00E+00	0.00E+00	0.00E+00	1.99E-05	9.68E-05	7.32E-06
Pu-239/240	6.56E-04	0.00E+00	0.00E+00	0.00E+00	2.44E-03	1.19E-02	9.00E-04
TOTALS	7.80E+00	1.80E-02	2.21E+02	3.01E-01	3.67E-02	1.20E+01	5.41E+00

Table C-30. Dose evaluation for 1957 operational releases (infant)

Report Date: 15-Apr-91
FileName: IDA13B.WK1

Operational Releases

Calendar Year..... 1957
Receptor Location..... Atomic City
Receptor Designation..... Infant
Release Location..... Midpoint between ICPP and TRA facilities

Input Parameters

5.00E-08 : Mesoscale Dispersion Factor (hr**2/m**3)
19.1 : Distance to Receptor (km)
16.4 : Annual Average Wind Speed Toward Receptor (km/hr)
1.00E+00 : Exposure Period (y) [8.77E+03 hours]
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

FileName: IDA13B.WK1

1957 Operational Releases

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surfac	Red Marrow
Immersion	2.42E-01	6.96E-01	2.68E-01	2.07E-01	1.93E-01	2.52E-01	1.95E-01
Inhalation	5.76E-01	8.82E-03	1.09E+01	1.47E+00	2.65E-01	7.43E-01	1.87E-01
Grnd Surface	2.24E+00	3.81E+01	2.46E+00	1.96E+00	1.83E+00	2.26E+00	2.00E+00
Ingestion	2.72E+01	3.72E-01	8.68E+02	4.89E-01	4.09E-01	9.80E+00	7.13E+00
TOTALS	3.03E+01	3.92E+01	8.82E+02	4.13E+00	2.69E+00	1.31E+01	9.51E+00

Table C-30. (continued).

Report Date: 15-Apr-91
Infant receptor

1957 Operational Releases

Source Term and Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
H-3	1.08E+05	5.51E+03	3.59E-12	3.14E-08	0.00E+00
Na-24	1.50E+01	3.66E+00	2.26E-15	1.98E-11	1.80E-03
Ar-41	1.83E+00	3.42E+04	1.43E-11	1.25E-07	0.00E+00
Cr-51	6.65E+02	1.27E-02	8.25E-18	7.24E-14	1.80E-03
Co-60	4.62E+04	9.62E-02	6.26E-17	5.49E-13	1.80E-03
Kr-85	9.40E+04	1.00E+05	6.52E-11	5.71E-07	0.00E+00
Kr-85m	4.48E+00	1.56E+03	8.46E-13	7.42E-09	0.00E+00
Kr-87	1.27E+00	5.59E+03	1.93E-12	1.69E-08	0.00E+00
Kr-88	2.84E+00	5.48E+03	2.68E-12	2.35E-08	0.00E+00
Rb-89	2.57E-01	4.48E+02	1.27E-14	1.11E-10	1.80E-03
Sr-89	1.21E+03	1.92E+01	1.25E-14	1.09E-10	1.80E-03
Sr-90	2.51E+05	3.57E+01	2.32E-14	2.04E-10	1.80E-03
Sr-91	9.50E+00	9.19E-01	5.49E-16	4.81E-12	1.80E-03
Y-91	1.40E+03	2.11E+01	1.37E-14	1.20E-10	1.80E-03
Y-91m	8.29E-01	5.01E-01	1.23E-16	1.08E-12	1.80E-03
Zr-95	1.54E+03	2.13E+01	1.39E-14	1.21E-10	1.80E-03
Tc-99m	6.02E+00	1.11E+00	6.32E-16	5.54E-12	1.80E-03
Ru-103	9.44E+02	1.53E+01	9.95E-15	8.72E-11	1.80E-03
Ru-106	8.84E+03	5.87E+00	3.82E-15	3.35E-11	1.80E-03
Te-132	7.82E+01	3.41E+01	2.20E-14	1.93E-10	1.80E-03
I-129 (ELE)	1.38E+11	4.30E-07	2.80E-22	2.45E-18	1.80E-02
I-129 (ORG)	1.38E+11	8.60E-07	5.60E-22	4.91E-18	1.80E-04
I-131	1.93E+02	1.40E+03	9.07E-13	7.95E-09	1.80E-02
I-132	2.30E+00	4.03E+03	1.85E-12	1.62E-08	1.80E-02
I-133	2.08E+01	4.41E+02	2.76E-13	2.42E-09	1.80E-02
I-135	6.61E+00	6.65E+00	3.83E-15	3.36E-11	1.80E-02
Xe-131m	2.84E+02	1.65E+03	1.07E-12	9.38E-09	0.00E+00
Xe-133	1.26E+02	4.63E+05	2.99E-10	2.62E-06	0.00E+00
Xe-133m	5.26E+01	9.16E+03	5.87E-12	5.15E-08	0.00E+00
Xe-135	9.11E+00	1.83E+04	1.09E-11	9.57E-08	0.00E+00
Xe-135m	2.56E-01	3.28E+03	9.12E-14	8.00E-10	0.00E+00
Xe-138	2.36E-01	1.83E+04	3.87E-13	3.39E-09	0.00E+00
Cs-134	1.81E+04	6.07E-01	3.95E-16	3.46E-12	1.80E-03
Cs-136	3.16E+02	3.17E+00	2.06E-15	1.80E-11	1.80E-03
Cs-137	2.65E+05	3.59E+01	2.34E-14	2.05E-10	1.80E-03
Cs-138	5.37E-01	4.04E+02	5.84E-14	5.12E-10	1.80E-03
Ba-139	1.39E+00	3.82E+01	1.39E-14	1.22E-10	1.80E-03
Ba-140	3.07E+02	5.85E+01	3.80E-14	3.33E-10	1.80E-03
Ce-141	7.80E+02	3.32E+01	2.16E-14	1.89E-10	1.80E-03
Ce-144	6.82E+03	5.40E+01	3.51E-14	3.08E-10	1.80E-03
Pr-143	3.25E+02	5.79E+01	3.76E-14	3.29E-10	1.80E-03
Pm-147	2.30E+04	1.40E+02	9.11E-14	7.99E-10	1.80E-03
Eu-154	7.71E+04	7.05E-02	4.59E-17	4.02E-13	1.80E-03
Hg-203	1.12E+03	1.43E+00	9.30E-16	8.15E-12	1.80E-03
Pu-238	7.69E+05	7.78E-02	5.06E-17	4.44E-13	1.80E-03
Pu-239	2.12E+08	9.36E-05	6.09E-20	5.34E-16	1.80E-03
Pu-239/240	2.12E+08	1.15E-02	7.48E-18	6.56E-14	1.80E-03
TOTAL		6.73E+05			

Table C-30. (continued).

Report Date: 15-Apr-91
Infant receptor

1957 Operational Releases

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.36E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02
Na-24	1.37E-04	3.00E-04	1.41E-04	1.32E-04	1.12E-04	1.25E-04	1.24E-04
Ar-41	9.48E-02	1.74E-01	1.05E-01	8.42E-02	7.96E-02	8.99E-02	8.49E-02
Cr-51	1.34E-07	1.64E-07	1.33E-07	1.28E-07	9.72E-08	1.34E-07	1.11E-07
Co-60	4.39E-03	5.76E-03	4.76E-03	3.99E-03	3.77E-03	4.08E-03	3.92E-03
Kr-85	7.29E-04	1.03E-01	7.88E-04	6.35E-04	5.91E-04	7.42E-04	6.49E-04
Kr-85m	6.91E-04	2.22E-03	7.64E-04	5.69E-04	5.21E-04	8.37E-04	5.46E-04
Kr-87	8.61E-03	3.18E-02	9.19E-03	7.61E-03	7.23E-03	8.27E-03	7.79E-03
Kr-88	4.11E-02	1.12E-01	4.41E-02	3.67E-02	3.49E-02	3.88E-02	3.74E-02
Rb-89	1.49E-04	3.20E-04	1.61E-04	1.36E-04	1.25E-04	1.40E-04	1.34E-04
Sr-89	1.40E-03	5.84E-02	1.51E-06	5.36E-03	1.13E-06	1.31E-06	1.22E-06
Sr-90	1.24E+00	2.88E+01	1.05E-01	1.06E-01	1.05E-01	9.53E+00	6.77E+00
Sr-91	7.45E-06	3.42E-05	7.04E-06	1.15E-05	5.25E-06	6.27E-06	5.70E-06
Y-91	1.05E-03	7.74E-02	4.67E-05	7.07E-03	3.51E-05	3.99E-05	3.77E-05
Y-91m	3.71E-07	5.35E-07	3.99E-07	3.64E-07	2.98E-07	3.69E-07	3.26E-07
Zr-95	3.86E-02	4.73E-02	3.85E-02	5.28E-02	2.92E-02	3.71E-02	3.18E-02
Tc-99m	8.14E-07	9.82E-07	1.03E-06	7.10E-07	5.76E-07	9.97E-07	5.92E-07
Ru-103	4.44E-03	4.65E-03	3.50E-03	9.86E-03	2.73E-03	3.32E-03	2.90E-03
Ru-106	2.69E-02	3.45E-01	6.18E-03	1.58E-01	4.98E-03	5.98E-03	5.85E-03
Te-132	3.47E-03	9.62E-03	1.11E-02	3.00E-03	2.63E-03	3.20E-03	2.85E-03
I-129 (ELE)	1.77E-07	3.00E-08	5.54E-06	6.64E-09	5.05E-09	1.14E-08	2.77E-09
I-129 (ORG)	3.73E-09	6.00E-10	1.17E-07	1.40E-10	1.01E-10	2.28E-10	5.59E-11
I-131	2.66E+01	1.33E+00	8.79E+02	6.40E-01	4.79E-01	6.49E-01	5.77E-01
I-132	9.50E-02	2.47E-01	2.65E-01	1.02E-01	7.19E-02	8.47E-02	7.76E-02
I-133	3.36E-02	1.66E-01	3.53E-01	2.09E-02	1.96E-02	2.41E-02	2.14E-02
I-135	2.78E-04	8.71E-04	4.71E-04	2.50E-04	2.27E-04	2.59E-04	2.44E-04
Xe-131m	4.59E-05	6.00E-04	4.72E-05	2.79E-05	2.36E-05	4.53E-05	1.89E-05
Xe-133	5.26E-02	1.74E-01	6.01E-02	3.89E-02	3.47E-02	6.87E-02	2.54E-02
Xe-133m	8.74E-04	7.10E-03	9.33E-04	6.75E-04	6.10E-04	9.50E-04	6.33E-04
Xe-135	1.36E-02	3.97E-02	1.48E-02	1.15E-02	1.06E-02	1.53E-02	1.17E-02
Xe-135m	1.96E-04	3.25E-04	2.12E-04	1.70E-04	1.59E-04	1.99E-04	1.74E-04
Xe-138	2.42E-03	4.99E-03	2.61E-03	2.15E-03	2.04E-03	2.34E-03	2.18E-03
Cs-134	1.05E-02	1.50E-02	1.13E-02	9.29E-03	9.23E-03	1.06E-02	1.00E-02
Cs-136	1.12E-03	1.28E-03	1.21E-03	9.92E-04	7.67E-04	1.10E-03	1.02E-03
Cs-137	1.78E+00	5.24E+00	1.93E+00	1.57E+00	1.48E+00	1.78E+00	1.68E+00
Cs-138	7.93E-04	1.88E-03	8.60E-04	7.47E-04	6.62E-04	7.49E-04	7.11E-04
Ba-139	6.06E-06	2.20E-04	3.30E-06	2.03E-05	2.27E-06	3.53E-06	2.37E-06
Ba-140	2.11E-02	8.64E-02	2.27E-02	1.87E-02	1.74E-02	2.06E-02	1.88E-02
Ce-141	1.25E-03	3.05E-03	1.09E-03	2.66E-03	7.07E-04	1.22E-03	7.01E-04
Ce-144	1.62E-01	2.03E+00	9.60E-03	1.12E+00	1.99E-02	3.65E-02	3.97E-02
Pr-143	4.11E-04	1.82E-02	8.19E-11	2.50E-03	6.11E-11	7.29E-11	6.64E-11
Pm-147	4.59E-03	1.55E-05	6.35E-06	3.53E-02	4.08E-06	7.36E-06	3.91E-06
Eu-154	2.77E-03	6.75E-03	3.04E-03	2.43E-03	2.37E-03	2.83E-03	2.43E-03
Hg-203	5.59E-04	2.21E-04	1.86E-04	3.16E-04	1.33E-04	1.92E-04	3.90E-04
Pu-238	4.80E-02	2.01E-03	1.96E-03	4.94E-02	2.19E-01	5.35E-01	6.97E-02
Pu-239	6.10E-05	2.35E-06	2.32E-06	5.63E-05	2.90E-04	7.09E-04	8.70E-05
Pu-240	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-30. (continued).

Report Date: 15-Apr-91
Infant receptor

1957 Operational Releases

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	5.39E-05	7.89E-05	5.73E-05	4.80E-05	4.58E-05	5.01E-05	4.89E-05
Ar-41	9.48E-02	1.74E-01	1.05E-01	8.42E-02	7.96E-02	8.99E-02	8.49E-02
Cr-51	1.30E-09	1.69E-09	1.41E-09	1.11E-09	1.03E-09	1.42E-09	1.15E-09
Co-60	8.13E-07	1.11E-06	9.01E-07	7.19E-07	6.82E-07	7.69E-07	7.26E-07
Kr-85	7.29E-04	1.03E-01	7.88E-04	6.35E-04	5.91E-04	7.42E-04	6.49E-04
Kr-85m	6.91E-04	2.22E-03	7.64E-04	5.69E-04	5.21E-04	8.37E-04	5.46E-04
Kr-87	8.61E-03	3.18E-02	9.19E-03	7.61E-03	7.23E-03	8.27E-03	7.79E-03
Kr-88+D	4.11E-02	1.12E-01	4.41E-02	3.67E-02	3.49E-02	3.88E-02	3.74E-02
Rb-89	1.44E-04	2.92E-04	1.57E-04	1.28E-04	1.21E-04	1.37E-04	1.30E-04
Sr-89	8.95E-09	5.46E-05	9.99E-09	7.91E-09	7.42E-09	8.66E-09	8.02E-09
Sr-90+D	0.00E+00	1.97E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	2.90E-06	6.68E-06	3.20E-06	2.55E-06	2.38E-06	2.84E-06	2.59E-06
Y-91	2.56E-07	6.27E-05	2.85E-07	2.28E-07	2.15E-07	2.44E-07	2.30E-07
Y-91m	3.30E-07	4.66E-07	3.59E-07	2.88E-07	2.68E-07	3.32E-07	2.94E-07
Zr-95+D	1.73E-04	2.41E-04	1.92E-04	1.52E-04	1.43E-04	1.70E-04	1.55E-04
Tc-99m	4.18E-07	5.30E-07	4.68E-07	3.40E-07	3.08E-07	5.32E-07	3.10E-07
Ru-103+D	2.41E-05	3.37E-05	2.60E-05	2.10E-05	1.95E-05	2.46E-05	2.15E-05
Ru-106+D	4.01E-06	4.89E-05	4.39E-06	3.52E-06	3.29E-06	4.01E-06	3.59E-06
Te-132+D	2.93E-04	4.76E-04	3.22E-04	2.57E-04	2.40E-04	2.90E-04	2.59E-04
I-129 (ELE)	1.17E-14	3.24E-14	1.16E-14	5.76E-15	4.61E-15	1.01E-14	2.19E-15
I-129 (ORG)	2.33E-14	6.49E-14	2.32E-14	1.15E-14	9.23E-15	2.02E-14	4.37E-15
I-131	1.73E-03	3.19E-03	1.88E-03	1.49E-03	1.38E-03	1.83E-03	1.53E-03
I-132	2.20E-02	3.60E-02	2.42E-02	1.94E-02	1.81E-02	2.14E-02	1.96E-02
I-133	8.47E-04	1.92E-03	9.24E-04	7.42E-04	6.93E-04	8.50E-04	7.56E-04
I-135	3.19E-05	5.24E-05	3.51E-05	2.84E-05	2.68E-05	3.04E-05	2.87E-05
Xe-131m	4.59E-05	6.00E-04	4.72E-05	2.79E-05	2.36E-05	4.53E-05	1.89E-05
Xe-133	5.26E-02	1.74E-01	6.01E-02	3.89E-02	3.47E-02	6.87E-02	2.54E-02
Xe-133m	8.74E-04	7.10E-03	9.33E-04	6.75E-04	6.10E-04	9.50E-04	6.33E-04
Xe-135	1.36E-02	3.97E-02	1.48E-02	1.15E-02	1.06E-02	1.53E-02	1.17E-02
Xe-135m	1.96E-04	3.25E-04	2.12E-04	1.70E-04	1.59E-04	1.99E-04	1.74E-04
Xe-138	2.42E-03	4.99E-03	2.61E-03	2.15E-03	2.04E-03	2.34E-03	2.18E-03
Cs-134	3.16E-06	4.62E-06	3.49E-06	2.78E-06	2.60E-06	3.12E-06	2.83E-06
Cs-136	2.32E-05	3.21E-05	2.57E-05	2.04E-05	1.91E-05	2.28E-05	2.06E-05
Cs-137+D	6.76E-05	1.22E-04	7.44E-05	5.92E-05	5.54E-05	6.71E-05	6.05E-05
Cs-138	7.42E-04	1.55E-03	8.12E-04	6.60E-04	6.25E-04	7.07E-04	6.72E-04
Ba-139	2.50E-06	9.99E-05	2.76E-06	2.07E-06	1.90E-06	2.95E-06	1.98E-06
Ba-140+D	5.72E-04	9.96E-04	6.23E-04	5.07E-04	4.78E-04	5.52E-04	5.16E-04
Ce-141	8.32E-06	2.57E-05	9.30E-06	6.71E-06	6.08E-06	1.05E-05	6.08E-06
Ce-144+D	9.39E-06	3.54E-04	1.03E-05	8.02E-06	7.50E-06	9.96E-06	7.74E-06
Pr-143	1.74E-12	7.77E-05	1.92E-12	1.52E-12	1.43E-12	1.70E-12	1.55E-12
Pm-147	1.67E-09	8.73E-06	1.88E-09	1.34E-09	1.21E-09	2.18E-09	1.16E-09
Eu-154	2.98E-07	4.63E-07	3.31E-07	2.62E-07	2.46E-07	2.92E-07	2.64E-07
Hg-203	1.07E-06	1.67E-06	1.16E-06	8.98E-07	8.29E-07	1.20E-06	9.07E-07
Pu-238	2.23E-11	1.55E-10	8.65E-12	5.97E-12	3.62E-12	1.06E-11	2.69E-12
Pu-239	2.50E-14	8.83E-14	2.06E-14	1.47E-14	1.25E-14	2.44E-14	1.19E-14
Pu-240	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	2.42E-01	6.96E-01	2.68E-01	2.07E-01	1.93E-01	2.52E-01	1.95E-01

Table C-30. (continued).

Report Date: 15-Apr-91
Infant receptor

1957 Operational Releases

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.01E-03	1.01E-03	1.01E-03	1.01E-03	1.01E-03	1.01E-03	1.01E-03
Na-24	2.97E-06	0.00E+00	0.00E+00	1.38E-05	0.00E+00	0.00E+00	2.44E-06
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	2.97E-09	0.00E+00	0.00E+00	2.29E-08	0.00E+00	0.00E+00	0.00E+00
Co-60	1.30E-05	0.00E+00	0.00E+00	1.13E-04	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	6.49E-07	0.00E+00	0.00E+00	4.38E-06	0.00E+00	0.00E+00	0.00E+00
Sr-89	6.40E-04	0.00E+00	0.00E+00	5.36E-03	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.43E-02	1.19E-03	1.19E-03	2.74E-03	1.19E-03	1.13E-01	8.11E-02
Sr-91	1.07E-06	0.00E+00	0.00E+00	5.93E-06	0.00E+00	0.00E+00	0.00E+00
Y-91	8.36E-04	0.00E+00	0.00E+00	7.03E-03	0.00E+00	0.00E+00	0.00E+00
Y-91m	5.28E-09	0.00E+00	0.00E+00	4.43E-08	0.00E+00	0.00E+00	0.00E+00
Zr-95	3.55E-03	2.71E-04	5.33E-04	2.26E-02	9.23E-04	1.77E-03	6.96E-04
Tc-99m	2.80E-08	0.00E+00	1.57E-07	7.44E-08	0.00E+00	0.00E+00	1.14E-08
Ru-103	1.12E-03	6.63E-05	1.12E-04	7.14E-03	1.79E-04	1.22E-04	1.02E-04
Ru-106	1.92E-02	2.94E-04	3.33E-04	1.53E-01	3.91E-04	3.91E-04	4.65E-04
Te-132	2.34E-04	0.00E+00	7.00E-03	1.92E-04	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.00E-10	2.01E-13	3.30E-09	3.73E-12	2.01E-13	5.46E-13	3.15E-13
I-129 (ORG)	2.01E-10	4.01E-13	6.60E-09	7.46E-12	4.01E-13	1.09E-12	6.31E-13
I-131	3.20E-01	1.39E-03	1.07E+01	3.25E-02	1.53E-03	2.05E-03	2.10E-03
I-132	9.46E-03	7.47E-04	1.70E-01	2.66E-02	1.32E-03	1.14E-03	1.14E-03
I-133	2.07E-03	0.00E+00	6.88E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	5.83E-06	0.00E+00	1.64E-04	8.49E-06	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.44E-05	2.23E-05	3.24E-05	4.25E-05	3.24E-05	3.24E-05	3.44E-05
Cs-136	2.14E-05	0.00E+00	1.80E-05	2.42E-05	0.00E+00	1.80E-05	1.99E-05
Cs-137	1.56E-03	1.18E-03	1.44E-03	2.39E-03	1.44E-03	1.56E-03	2.04E-03
Cs-138	7.12E-06	0.00E+00	0.00E+00	4.77E-05	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.08E-06	0.00E+00	0.00E+00	1.79E-05	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.89E-04	0.00E+00	0.00E+00	3.31E-04	0.00E+00	4.68E-04	2.52E-04
Ce-141	2.54E-04	0.00E+00	0.00E+00	1.88E-03	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.42E-01	4.68E-04	5.60E-04	1.11E+00	9.93E-03	1.73E-02	1.98E-02
Pr-143	3.80E-04	0.00E+00	0.00E+00	2.50E-03	0.00E+00	0.00E+00	0.00E+00
Pm-147	4.29E-03	0.00E+00	0.00E+00	3.53E-02	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.65E-05	0.00E+00	0.00E+00	1.84E-05	1.02E-04	1.21E-04	2.60E-05
Hg-203	8.37E-06	0.00E+00	0.00E+00	5.28E-06	0.00E+00	0.00E+00	5.28E-06
Pu-238	4.67E-02	1.89E-03	1.89E-03	4.93E-02	2.12E-01	5.19E-01	6.74E-02
Pu-239	5.93E-05	2.24E-06	2.24E-06	5.62E-05	2.81E-04	6.87E-04	8.43E-05
Pu-239/240	7.29E-03	2.76E-04	2.76E-04	6.90E-03	3.45E-02	8.44E-02	1.04E-02
TOTALS	5.76E-01	8.82E-03	1.09E+01	1.47E+00	2.65E-01	7.43E-01	1.87E-01

Table C-30. (continued).

Report Date: 15-Apr-91
Infant receptor

1957 Operational Releases

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	7.86E-05	2.21E-04	8.35E-05	6.98E-05	6.67E-05	7.31E-05	7.13E-05
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.22E-07	1.63E-07	1.32E-07	1.04E-07	9.62E-08	1.32E-07	1.07E-07
Co-60	4.29E-03	5.76E-03	4.76E-03	3.82E-03	3.61E-03	4.08E-03	3.85E-03
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.05E-06	2.81E-05	4.43E-06	3.60E-06	3.41E-06	3.86E-06	3.66E-06
Sr-89	1.35E-06	5.83E-02	1.50E-06	1.19E-06	1.12E-06	1.30E-06	1.21E-06
Sr-90+D	0.00E+00	2.87E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	3.48E-06	2.76E-05	3.84E-06	3.06E-06	2.87E-06	3.42E-06	3.11E-06
Y-91	4.17E-05	7.74E-02	4.64E-05	3.71E-05	3.49E-05	3.97E-05	3.74E-05
Y-91m	3.64E-08	6.87E-08	3.96E-08	3.18E-08	2.97E-08	3.67E-08	3.25E-08
Zr-95+D	3.40E-02	4.68E-02	3.78E-02	3.00E-02	2.81E-02	3.34E-02	3.05E-02
Tc-99m	3.68E-07	4.52E-07	4.07E-07	2.96E-07	2.68E-07	4.65E-07	2.71E-07
Ru-103+D	3.08E-03	4.52E-03	3.33E-03	2.67E-03	2.49E-03	3.14E-03	2.74E-03
Ru-106+D	4.64E-03	3.44E-01	5.06E-03	4.06E-03	3.80E-03	4.66E-03	4.15E-03
Te-132+D	2.93E-03	9.14E-03	3.23E-03	2.55E-03	2.39E-03	2.91E-03	2.59E-03
I-129 (ELE)	1.24E-08	2.96E-08	1.22E-08	6.01E-09	4.73E-09	1.05E-08	2.26E-09
I-129 (ORG)	2.47E-10	5.93E-10	2.44E-10	1.20E-10	9.47E-11	2.10E-10	4.51E-11
I-131	4.63E-01	1.21E+00	5.02E-01	3.97E-01	3.69E-01	4.90E-01	4.09E-01
I-132	6.36E-02	2.10E-01	7.02E-02	5.58E-02	5.25E-02	6.22E-02	5.69E-02
I-133	2.31E-02	1.64E-01	2.52E-02	2.02E-02	1.89E-02	2.32E-02	2.06E-02
I-135	2.40E-04	8.19E-04	2.63E-04	2.13E-04	2.00E-04	2.28E-04	2.15E-04
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	7.38E-03	1.29E-02	8.13E-03	6.49E-03	6.07E-03	7.29E-03	6.59E-03
Cs-136	9.10E-04	1.25E-03	1.01E-03	7.99E-04	7.48E-04	9.01E-04	8.08E-04
Cs-137+D	1.60E+00	5.10E+00	1.76E+00	1.41E+00	1.31E+00	1.59E+00	1.43E+00
Cs-138	4.41E-05	3.28E-04	4.80E-05	3.92E-05	3.71E-05	4.20E-05	3.96E-05
Ba-139	4.88E-07	1.20E-04	5.39E-07	4.01E-07	3.67E-07	5.78E-07	3.82E-07
Ba-140+D	2.02E-02	8.54E-02	2.21E-02	1.79E-02	1.69E-02	1.95E-02	1.81E-02
Ce-141	9.67E-04	3.03E-03	1.08E-03	7.73E-04	7.01E-04	1.21E-03	6.95E-04
Ce-144+D	7.99E-03	2.03E+00	8.79E-03	6.73E-03	6.25E-03	8.66E-03	6.34E-03
Pr-143	7.25E-11	1.82E-02	8.00E-11	6.38E-11	5.97E-11	7.12E-11	6.49E-11
Pm-147	5.62E-06	6.80E-06	6.34E-06	4.52E-06	4.08E-06	7.36E-06	3.91E-06
Eu-154	2.75E-03	6.75E-03	3.04E-03	2.41E-03	2.26E-03	2.70E-03	2.41E-03
Hg-203	1.70E-04	2.19E-04	1.85E-04	1.43E-04	1.32E-04	1.91E-04	1.44E-04
Pu-238	7.19E-06	6.01E-05	1.26E-06	1.02E-06	2.20E-07	1.72E-06	3.78E-07
Pu-239	4.62E-09	3.42E-08	1.42E-09	1.09E-09	5.92E-10	1.80E-09	6.81E-10
Pu-239/240	5.67E-07	4.20E-06	1.74E-07	1.34E-07	7.28E-08	2.21E-07	8.37E-08
TOTALS	2.24E+00	3.81E+01	2.46E+00	1.96E+00	1.83E+00	2.26E+00	2.00E+00

Table C-30. (continued).

Report Date: 15-Apr-91
Infant receptor

1957 Operational Releases

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
H-3	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02
Na-24	1.18E-06	0.00E+00	0.00E+00	8.09E-07	0.00E+00	1.43E-06	1.18E-06
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	7.79E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.64E-09
Co-60	8.86E-05	0.00E+00	0.00E+00	6.48E-05	1.64E-04	0.00E+00	6.82E-05
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.08E-38	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	7.61E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.22E+00	1.04E-01	1.04E-01	1.04E-01	1.04E-01	9.41E+00	6.69E+00
Sr-91	7.19E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	1.70E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91m	7.10E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	7.82E-04	5.16E-05	2.98E-05	5.16E-05	8.60E-05	1.80E-03	4.92E-04
Tc-99m	7.70E-11	0.00E+00	3.98E-10	0.00E+00	0.00E+00	0.00E+00	2.95E-11
Ru-103	2.15E-04	2.74E-05	3.08E-05	3.08E-05	4.20E-05	3.92E-05	4.48E-05
Ru-106	3.05E-03	7.55E-04	7.89E-04	7.89E-04	7.89E-04	9.27E-04	1.23E-03
Te-132	1.75E-05	0.00E+00	5.22E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129 (ELE)	1.64E-07	3.28E-10	5.53E-06	6.25E-10	3.14E-10	8.51E-10	5.08E-10
I-129 (ORG)	3.28E-09	6.57E-12	1.11E-07	1.25E-11	6.27E-12	1.70E-11	1.02E-11
I-131	2.58E+01	1.10E-01	8.68E+02	2.08E-01	1.08E-01	1.55E-01	1.64E-01
I-132	2.00E-07	1.67E-08	4.57E-06	3.34E-08	2.67E-08	2.34E-08	2.67E-08
I-133	7.60E-03	0.00E+00	2.58E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.56E-07	0.00E+00	8.58E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.13E-03	2.13E-03	3.13E-03	2.75E-03	3.13E-03	3.25E-03	3.38E-03
Cs-136	1.70E-04	0.00E+00	1.55E-04	1.49E-04	0.00E+00	1.55E-04	1.70E-04
Cs-137	1.78E-01	1.42E-01	1.69E-01	1.60E-01	1.69E-01	1.87E-01	2.40E-01
Cs-138	1.63E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	4.26E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.68E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	2.32E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.23E-02	2.46E-04	2.46E-04	2.46E-04	3.68E-03	1.06E-02	1.35E-02
Pr-143	3.10E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-147	2.91E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.55E-06	0.00E+00	0.00E+00	0.00E+00	2.39E-06	2.90E-06	7.00E-07
Hg-203	3.80E-04	0.00E+00	0.00E+00	1.67E-04	0.00E+00	0.00E+00	2.39E-04
Pu-238	1.34E-03	6.18E-05	6.18E-05	6.29E-05	6.71E-03	1.65E-02	2.27E-03
Pu-239	1.74E-06	7.45E-08	7.45E-08	7.45E-08	8.82E-06	2.23E-05	2.73E-06
Pu-239/240	2.14E-04	9.15E-06	9.15E-06	9.15E-06	1.08E-03	2.75E-03	3.36E-04
TOTALS	2.72E+01	3.72E-01	8.68E+02	4.89E-01	4.09E-01	9.80E+00	7.13E+00

Table C-31. Dose evaluation for the SL-1 Accident (adult)

Report Date: 27-Jul-91
File Name: ACC8A.WK1

Episodic Release

Episode Description..... SL-1 Accident
Date..... 1/3/61
Receptor Location..... Atomic City
Receptor Designation..... Adult
Release Location..... SL-1

Input Parameters

8.00E-11 : Episodic Mesoscale Dispersion Factor (hr/m**3)
1.60E+01 : Exposure Period (hr)
9.59E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 27-Jul-91

1/3/61 Episodic Release
SL-1 Accident

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	7.11E-02	1.82E-01	7.67E-02	6.31E-02	5.98E-02	6.84E-02	6.42E-02
Inhalation	6.70E-01	0.00E+00	6.86E+00	3.58E+00	2.47E-06	6.16E-01	8.54E-02
Grnd Surface	1.33E+00	1.18E+01	1.46E+00	1.16E+00	1.08E+00	1.34E+00	1.18E+00
Ingestion	1.11E+00	0.00E+00	3.10E+01	5.40E-02	0.00E+00	5.13E-01	2.55E-01
TOTALS	3.18E+00	1.20E+01	3.94E+01	4.86E+00	1.14E+00	2.54E+00	1.58E+00

Table C-31. (continued).

Report Date: 27-Jul-91
Adult receptor

1/3/61 Episodic Release
SL-1 Accident

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Br-84	5.30E-01	1.90E-02	9.50E-14	1.52E-12	1.80E-03
Kr-85m	4.48E+00	3.55E+01	1.78E-10	2.84E-09	0.00E+00
Kr-87	1.27E+00	2.52E+02	1.26E-09	2.02E-08	0.00E+00
Kr-88	2.84E+00	2.14E+02	1.07E-09	1.71E-08	0.00E+00
Rb-89	2.53E-01	5.46E-02	2.73E-13	4.37E-12	1.80E-03
Sr-89	1.21E+03	3.25E+00	1.63E-11	2.60E-10	1.80E-03
Sr-90	2.50E+05	1.00E-01	5.00E-13	8.00E-12	1.80E-03
Sr-91	9.50E+00	4.13E-03	2.07E-14	3.30E-13	1.80E-03
Sr-92	2.71E+00	1.21E-02	6.05E-14	9.68E-13	1.80E-03
Y-91	1.40E+03	1.95E+01	9.75E-11	1.56E-09	1.80E-03
Y-92	3.54E+00	1.19E-02	5.95E-14	9.52E-13	1.80E-03
Y-93	1.02E+01	2.06E-02	1.03E-13	1.65E-12	1.80E-03
Zr-95	1.54E+03	2.13E+01	1.07E-10	1.70E-09	1.80E-03
Zr-97	1.69E+01	3.28E-02	1.64E-13	2.62E-12	1.80E-03
Nb-96	2.34E+01	1.51E-05	7.55E-17	1.21E-15	1.80E-03
Mo-99	6.60E+01	3.66E+00	1.83E-11	2.93E-10	1.80E-03
Ru-103	9.44E+02	1.01E+01	5.05E-11	8.08E-10	1.80E-03
Ru-105	4.44E+00	7.06E-03	3.53E-14	5.65E-13	1.80E-03
Ru-106	8.84E+03	7.17E-01	3.59E-12	5.74E-11	1.80E-03
Sb-129	4.32E+00	4.67E-03	2.34E-14	3.74E-13	1.80E-03
Te-131	4.20E-01	7.18E-02	3.59E-13	5.74E-12	1.80E-03
Te-131m	3.00E+01	2.74E-02	1.37E-13	2.19E-12	1.80E-03
Te-132	7.80E+01	3.49E+00	1.75E-11	2.79E-10	1.80E-03
Te-133m	9.20E-01	5.18E-02	2.59E-13	4.14E-12	1.80E-03
Te-134	7.00E-01	1.18E-01	5.90E-13	9.44E-12	1.80E-03
I-131 (ELE)	1.93E+02	8.00E+01	4.00E-10	6.40E-09	1.80E-02
I-132	2.30E+00	4.95E+01	2.48E-10	3.96E-09	1.80E-02
I-133	2.08E+01	1.33E+00	6.65E-12	1.06E-10	1.80E-02
I-134	8.77E-01	1.54E+00	7.70E-12	1.23E-10	1.80E-02
I-135	6.61E+00	4.07E-01	2.04E-12	3.26E-11	1.80E-02
Xe-129m	2.13E+02	7.65E-05	3.83E-16	6.12E-15	0.00E+00
Xe-135	9.08E+00	2.09E+01	1.05E-10	1.67E-09	0.00E+00
Xe-135m	2.56E-01	3.90E+01	1.95E-10	3.12E-09	0.00E+00
Xe-138	2.36E-01	3.01E+02	1.51E-09	2.41E-08	0.00E+00
Cs-137	2.65E+05	5.00E-01	2.50E-12	4.00E-11	1.80E-03
Cs-138	5.37E-01	1.63E-01	8.15E-13	1.30E-11	1.80E-03
Ba-139	1.38E+00	1.08E-01	5.40E-13	8.64E-12	1.80E-03
Ba-140	3.07E+02	1.56E+01	7.80E-11	1.25E-09	1.80E-03
Ba-141	3.00E-01	6.98E-02	3.49E-13	5.58E-12	1.80E-03
Ba-142	1.80E-01	2.30E-02	1.15E-13	1.84E-12	1.80E-03
La-141	3.93E+00	4.13E-02	2.07E-13	3.30E-12	1.80E-03
La-142	1.59E+00	8.99E-02	4.50E-13	7.19E-12	1.80E-03
Ce-141	7.80E+02	1.86E+01	9.30E-11	1.49E-09	1.80E-03
Ce-143	3.30E+01	6.30E-01	3.15E-12	5.04E-11	1.80E-03
Ce-144	6.82E+03	1.10E+01	5.50E-11	8.80E-10	1.80E-03
Pr-143	3.25E+02	1.66E+01	8.30E-11	1.33E-09	1.80E-03
Pr-144	2.90E-01	1.10E+01	5.50E-11	8.80E-10	1.80E-03
TOTAL		1.13E+03			

Table C-31. (continued).

Report Date: 27-Jul-91
Adult receptor

1/3/61 Episodic Release
SL-1 Accident

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	1.99E-06	4.97E-06	1.99E-06	2.52E-06	1.58E-06	1.75E-06	1.70E-06
Kr-85m	2.65E-04	8.48E-04	2.92E-04	2.18E-04	1.99E-04	3.21E-04	2.09E-04
Kr-87	1.03E-02	3.79E-02	1.10E-02	9.08E-03	8.62E-03	9.86E-03	9.28E-03
Kr-88	2.99E-02	8.16E-02	3.21E-02	2.67E-02	2.54E-02	2.82E-02	2.72E-02
Rb-89	5.99E-06	1.26E-05	6.35E-06	6.22E-06	4.90E-06	5.53E-06	5.27E-06
Sr-89	1.70E-02	1.39E-01	3.58E-06	7.73E-02	2.68E-06	3.10E-06	2.89E-06
Sr-90	4.72E-02	1.12E+00	0.00E+00	8.44E-02	0.00E+00	4.59E-01	2.01E-01
Sr-91	8.82E-07	2.35E-06	4.83E-07	2.86E-06	3.60E-07	4.30E-07	3.91E-07
Sr-92	1.71E-06	1.59E-06	1.10E-06	4.69E-06	8.38E-07	9.47E-07	8.96E-07
Y-91	6.72E-02	1.00E+00	6.05E-04	5.54E-01	4.56E-04	5.18E-04	4.88E-04
Y-92	7.70E-07	5.49E-06	2.26E-07	4.20E-06	1.69E-07	1.97E-07	1.83E-07
Y-93	3.53E-06	1.80E-05	2.15E-07	1.49E-05	1.64E-07	1.95E-07	1.77E-07
Zr-95	5.28E-01	6.60E-01	5.33E-01	4.23E-01	3.96E-01	1.08E+00	5.09E-01
Zr-97	1.79E-05	5.26E-05	8.58E-06	4.46E-05	6.40E-06	7.67E-06	6.96E-06
Nb-96	9.39E-09	1.24E-08	7.67E-09	1.29E-08	5.72E-09	6.78E-09	6.19E-09
Mo-99	1.32E-03	5.25E-03	2.85E-04	4.72E-03	2.09E-04	2.61E-04	2.25E-04
Ru-103	3.94E-02	4.22E-02	3.11E-02	7.07E-02	2.32E-02	2.93E-02	2.55E-02
Ru-105	6.36E-07	1.38E-06	4.55E-07	1.50E-06	3.37E-07	4.16E-07	3.68E-07
Ru-106	3.66E-02	5.90E-01	8.67E-03	2.10E-01	6.51E-03	7.99E-03	7.12E-03
Sb-129	7.13E-07	1.99E-06	5.62E-07	1.63E-06	4.21E-07	4.97E-07	4.57E-07
Te-131	2.04E-06	6.72E-06	1.32E-05	2.94E-06	1.20E-06	1.58E-06	1.28E-06
Te-131m	2.13E-05	2.37E-05	2.66E-04	2.53E-05	7.71E-06	9.29E-06	8.30E-06
Te-132	6.80E-03	1.39E-02	6.90E-02	5.75E-03	3.81E-03	4.63E-03	4.13E-03
Te-133m	8.09E-06	1.42E-05	4.54E-05	8.89E-06	5.43E-06	6.42E-06	5.85E-06
Te-134	6.13E-06	7.53E-06	2.38E-05	6.51E-06	4.21E-06	5.39E-06	4.58E-06
I-131 (ELE)	1.48E+00	9.80E-01	3.81E+01	3.21E-01	2.98E-01	3.96E-01	3.30E-01
I-132	2.22E-02	6.03E-02	4.70E-02	2.22E-02	1.73E-02	2.05E-02	1.87E-02
I-133	1.75E-03	7.32E-03	2.43E-02	9.23E-04	8.63E-04	1.06E-03	9.43E-04
I-134	4.15E-04	1.09E-03	5.75E-04	4.17E-04	3.33E-04	3.90E-04	3.59E-04
I-135	2.98E-04	8.46E-04	1.27E-03	2.85E-04	2.20E-04	2.51E-04	2.37E-04
Xe-129m	8.16E-11	6.71E-10	8.37E-11	5.09E-11	4.37E-11	8.02E-11	3.61E-11
Xe-135	2.38E-04	6.94E-04	2.59E-04	2.00E-04	1.85E-04	2.67E-04	2.04E-04
Xe-135m	7.65E-04	1.27E-03	8.29E-04	6.65E-04	6.19E-04	7.75E-04	6.79E-04
Xe-138	1.72E-02	3.54E-02	1.85E-02	1.53E-02	1.45E-02	1.66E-02	1.55E-02
Cs-137	3.69E-01	9.90E-01	3.98E-01	3.29E-01	2.55E-01	3.64E-01	3.34E-01
Cs-138	2.11E-05	4.79E-05	2.19E-05	2.52E-05	1.69E-05	1.91E-05	1.81E-05
Ba-139	1.54E-06	1.56E-05	2.34E-07	7.88E-06	1.61E-07	2.51E-07	1.68E-07
Ba-140	9.42E-02	3.24E-01	8.51E-02	7.64E-02	6.51E-02	8.60E-02	7.54E-02
Ba-141	3.43E-06	9.27E-06	3.32E-06	5.00E-06	2.47E-06	3.11E-06	2.68E-06
Ba-142	1.06E-06	2.12E-06	1.11E-06	1.23E-06	8.25E-07	9.89E-07	8.88E-07
La-141	1.83E-06	1.27E-05	1.33E-07	7.71E-06	2.57E-06	1.14E-07	1.08E-07
La-142	1.65E-05	3.24E-05	1.59E-05	2.09E-05	1.27E-05	1.40E-05	1.35E-05
Ce-141	2.63E-02	2.40E-02	8.56E-03	9.60E-02	5.56E-03	9.61E-03	5.51E-03
Ce-143	2.00E-04	5.13E-04	4.70E-05	7.13E-04	3.33E-05	4.56E-05	3.55E-05
Ce-144	3.64E-01	5.80E+00	2.51E-02	2.47E+00	1.79E-02	2.48E-02	1.81E-02
Pr-143	1.58E-02	7.35E-02	3.30E-10	6.11E-02	2.46E-10	2.94E-10	2.68E-10
Pr-144	5.40E-05	1.24E-03	2.00E-05	3.12E-04	1.56E-05	1.76E-05	1.68E-05

Table C-31. (continued).

Report Date: 27-Jul-91
Adult receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	1.77E-06	4.05E-06	1.89E-06	1.57E-06	1.50E-06	1.66E-06	1.61E-06
Kr-85m	2.65E-04	8.48E-04	2.92E-04	2.18E-04	1.99E-04	3.21E-04	2.09E-04
Kr-87	1.03E-02	3.79E-02	1.10E-02	9.08E-03	8.62E-03	9.86E-03	9.28E-03
Kr-88+D	2.99E-02	8.16E-02	3.21E-02	2.67E-02	2.54E-02	2.82E-02	2.72E-02
Rb-89	5.68E-06	1.15E-05	6.17E-06	5.03E-06	4.77E-06	5.38E-06	5.13E-06
Sr-89	2.13E-08	1.30E-04	2.37E-08	1.88E-08	1.76E-08	2.06E-08	1.91E-08
Sr-90+D	0.00E+00	7.74E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.99E-07	4.58E-07	2.19E-07	1.75E-07	1.64E-07	1.95E-07	1.78E-07
Sr-92	7.65E-07	1.15E-06	8.43E-07	6.80E-07	6.42E-07	7.26E-07	6.86E-07
Y-91	3.33E-06	8.13E-04	3.70E-06	2.95E-06	2.79E-06	3.17E-06	2.99E-06
Y-92	1.43E-07	1.47E-06	1.58E-07	1.26E-07	1.18E-07	1.38E-07	1.28E-07
Y-93	9.06E-08	1.90E-06	9.86E-08	7.98E-08	7.53E-08	8.89E-08	8.12E-08
Zr-95+D	2.43E-03	3.39E-03	2.70E-03	2.14E-03	2.01E-03	2.38E-03	2.18E-03
Zr-97+D	2.46E-06	5.99E-06	2.70E-06	2.16E-06	2.02E-06	2.42E-06	2.20E-06
Nb-96	1.76E-09	2.59E-09	1.96E-09	1.56E-09	1.46E-09	1.72E-09	1.57E-09
Mo-99+D	2.67E-05	1.34E-04	2.95E-05	2.33E-05	2.17E-05	2.70E-05	2.36E-05
Ru-103+D	2.23E-04	3.13E-04	2.41E-04	1.94E-04	1.81E-04	2.28E-04	1.99E-04
Ru-105+D	2.61E-07	5.33E-07	2.88E-07	2.28E-07	2.13E-07	2.62E-07	2.33E-07
Ru-106+D	6.87E-06	8.37E-05	7.52E-06	6.03E-06	5.63E-06	6.87E-06	6.15E-06
Sb-129+D	3.33E-07	7.34E-07	3.67E-07	2.94E-07	2.75E-07	3.24E-07	2.98E-07
Te-131	1.41E-06	5.34E-06	1.56E-06	1.22E-06	1.13E-06	1.49E-06	1.21E-06
Te-131m+D	1.98E-06	3.18E-06	2.19E-06	1.74E-06	1.63E-06	1.95E-06	1.76E-06
Te-132+D	4.25E-04	6.90E-04	4.67E-04	3.73E-04	3.48E-04	4.20E-04	3.76E-04
Te-133m+D	5.91E-06	1.09E-05	6.51E-06	5.20E-06	4.88E-06	5.77E-06	5.25E-06
Te-134	4.78E-06	6.91E-06	5.25E-06	4.13E-06	3.85E-06	4.93E-06	4.19E-06
I-131 (ELE)	1.39E-03	2.57E-03	1.51E-03	1.20E-03	1.11E-03	1.47E-03	1.23E-03
I-132	5.37E-03	8.80E-03	5.91E-03	4.74E-03	4.43E-03	5.24E-03	4.79E-03
I-133	3.72E-05	8.45E-05	4.06E-05	3.26E-05	3.04E-05	3.74E-05	3.32E-05
I-134	1.94E-04	3.24E-04	2.15E-04	1.71E-04	1.60E-04	1.88E-04	1.73E-04
I-135	3.10E-05	5.09E-05	3.40E-05	2.75E-05	2.60E-05	2.95E-05	2.78E-05
Xe-129m	8.16E-11	6.71E-10	8.37E-11	5.09E-11	4.37E-11	8.02E-11	3.61E-11
Xe-135	2.38E-04	6.94E-04	2.59E-04	2.00E-04	1.85E-04	2.67E-04	2.04E-04
Xe-135m	7.65E-04	1.27E-03	8.29E-04	6.65E-04	6.19E-04	7.75E-04	6.79E-04
Xe-138	1.72E-02	3.54E-02	1.85E-02	1.53E-02	1.45E-02	1.66E-02	1.55E-02
Cs-137+D	1.32E-05	2.38E-05	1.45E-05	1.16E-05	1.08E-05	1.31E-05	1.18E-05
Cs-138	1.89E-05	3.95E-05	2.07E-05	1.68E-05	1.59E-05	1.80E-05	1.71E-05
Ba-139	1.77E-07	7.09E-06	1.96E-07	1.47E-07	1.35E-07	2.10E-07	1.41E-07
Ba-140+D	2.15E-03	3.73E-03	2.34E-03	1.90E-03	1.79E-03	2.07E-03	1.93E-03
Ba-141	2.92E-06	8.02E-06	3.20E-06	2.55E-06	2.38E-06	2.99E-06	2.58E-06
Ba-142	9.80E-07	1.99E-06	1.09E-06	8.60E-07	8.08E-07	9.67E-07	8.68E-07
La-141	8.40E-08	2.90E-06	9.23E-08	7.46E-08	7.08E-08	7.95E-08	7.53E-08
La-142	1.29E-05	2.21E-05	1.37E-05	1.15E-05	1.09E-05	1.21E-05	1.16E-05
Ce-141	6.55E-05	2.02E-04	7.31E-05	5.28E-05	4.78E-05	8.23E-05	4.78E-05
Ce-143	7.58E-06	2.68E-05	8.27E-06	6.38E-06	5.92E-06	7.99E-06	6.38E-06
Ce-144+D	2.68E-05	1.01E-03	2.93E-05	2.29E-05	2.14E-05	2.85E-05	2.21E-05
Pr-143	6.99E-12	3.13E-04	7.74E-12	6.15E-12	5.75E-12	6.86E-12	6.25E-12
Pr-144	1.80E-05	9.90E-04	1.94E-05	1.60E-05	1.51E-05	1.71E-05	1.63E-05
TOTALS	7.11E-02	1.82E-01	7.67E-02	6.31E-02	5.98E-02	6.84E-02	6.42E-02

Table C-31. (continued).

Report Date: 27-Jul-91
Adult receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	1.27E-07	0.00E+00	0.00E+00	8.60E-07	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.55E-07	0.00E+00	0.00E+00	1.05E-06	0.00E+00	0.00E+00	0.00E+00
Sr-89	9.23E-03	0.00E+00	0.00E+00	7.73E-02	0.00E+00	0.00E+00	0.00E+00
Sr-90	9.97E-03	0.00E+00	0.00E+00	8.44E-02	0.00E+00	0.00E+00	0.00E+00
Sr-91	4.44E-07	0.00E+00	0.00E+00	2.47E-06	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.15E-07	0.00E+00	0.00E+00	3.81E-06	0.00E+00	0.00E+00	0.00E+00
Y-91	6.58E-02	0.00E+00	0.00E+00	5.54E-01	0.00E+00	0.00E+00	0.00E+00
Y-92	5.66E-07	0.00E+00	0.00E+00	4.02E-06	0.00E+00	0.00E+00	0.00E+00
Y-93	3.32E-06	0.00E+00	0.00E+00	1.47E-05	0.00E+00	0.00E+00	0.00E+00
Zr-95	3.10E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.05E-01	7.84E-02
Zr-97	1.01E-05	0.00E+00	0.00E+00	3.77E-05	0.00E+00	0.00E+00	0.00E+00
Nb-96	2.32E-09	0.00E+00	0.00E+00	6.83E-09	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.01E-03	0.00E+00	0.00E+00	4.49E-03	0.00E+00	0.00E+00	0.00E+00
Ru-103	6.04E-03	0.00E+00	0.00E+00	4.57E-02	0.00E+00	0.00E+00	0.00E+00
Ru-105	2.22E-07	0.00E+00	0.00E+00	1.14E-06	0.00E+00	0.00E+00	0.00E+00
Ru-106	2.42E-02	0.00E+00	0.00E+00	2.04E-01	0.00E+00	0.00E+00	0.00E+00
Sb-129	2.04E-07	0.00E+00	0.00E+00	1.18E-06	0.00E+00	0.00E+00	0.00E+00
Te-131	5.45E-07	0.00E+00	1.16E-05	1.65E-06	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.16E-05	0.00E+00	2.52E-04	1.70E-05	0.00E+00	0.00E+00	0.00E+00
Te-132	2.06E-03	0.00E+00	6.16E-02	1.69E-03	0.00E+00	0.00E+00	0.00E+00
Te-133m	1.51E-06	0.00E+00	3.82E-05	3.10E-06	0.00E+00	0.00E+00	0.00E+00
Te-134	9.05E-07	0.00E+00	1.81E-05	1.99E-06	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	1.96E-01	0.00E+00	6.75E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	1.25E-03	0.00E+00	2.39E-02	3.80E-03	0.00E+00	0.00E+00	0.00E+00
I-133	5.51E-04	0.00E+00	1.84E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.30E-05	0.00E+00	1.30E-04	6.14E-05	0.00E+00	0.00E+00	0.00E+00
I-135	3.43E-05	0.00E+00	9.68E-04	5.00E-05	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	1.23E-03	0.00E+00	1.11E-03	1.27E-03	0.00E+00	1.11E-03	1.19E-03
Cs-138	1.10E-06	0.00E+00	0.00E+00	7.38E-06	0.00E+00	0.00E+00	0.00E+00
Ba-139	1.33E-06	0.00E+00	0.00E+00	7.71E-06	0.00E+00	0.00E+00	0.00E+00
Ba-140	4.31E-03	0.00E+00	0.00E+00	7.54E-03	0.00E+00	1.07E-02	5.74E-03
Ba-141	3.96E-07	0.00E+00	0.00E+00	2.36E-06	0.00E+00	0.00E+00	0.00E+00
Ba-142	6.35E-08	0.00E+00	0.00E+00	3.53E-07	0.00E+00	0.00E+00	0.00E+00
La-141	1.71E-06	0.00E+00	0.00E+00	7.60E-06	2.47E-06	0.00E+00	0.00E+00
La-142	1.52E-06	0.00E+00	0.00E+00	7.59E-06	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.21E-02	0.00E+00	0.00E+00	8.99E-02	0.00E+00	0.00E+00	0.00E+00
Ce-143	1.55E-04	0.00E+00	0.00E+00	6.77E-04	0.00E+00	0.00E+00	0.00E+00
Ce-144	2.95E-01	0.00E+00	0.00E+00	2.45E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	9.30E-03	0.00E+00	0.00E+00	6.11E-02	0.00E+00	0.00E+00	0.00E+00
Pr-144	3.54E-05	0.00E+00	0.00E+00	2.95E-04	0.00E+00	0.00E+00	0.00E+00
TOTALS	6.70E-01	0.00E+00	6.86E+00	3.58E+00	2.47E-06	6.16E-01	8.54E-02

Table C-31. (continued).

Report Date: 27-Jul-91
Adult receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	9.56E-08	9.20E-07	1.03E-07	8.48E-08	8.06E-08	9.02E-08	8.66E-08
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.57E-07	1.09E-06	1.72E-07	1.40E-07	1.32E-07	1.49E-07	1.42E-07
Sr-89	3.20E-06	1.39E-01	3.55E-06	2.82E-06	2.66E-06	3.08E-06	2.87E-06
Sr-90+D	0.00E+00	1.12E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	2.39E-07	1.89E-06	2.64E-07	2.10E-07	1.97E-07	2.35E-07	2.14E-07
Sr-92	2.33E-07	4.46E-07	2.58E-07	2.07E-07	1.96E-07	2.21E-07	2.09E-07
Y-91	5.41E-04	1.00E+00	6.02E-04	4.81E-04	4.53E-04	5.15E-04	4.85E-04
Y-92	6.09E-08	4.02E-06	6.74E-08	5.38E-08	5.05E-08	5.88E-08	5.46E-08
Y-93	1.07E-07	1.61E-05	1.17E-07	9.43E-08	8.88E-08	1.06E-07	9.58E-08
Zr-95+D	4.78E-01	6.56E-01	5.30E-01	4.21E-01	3.94E-01	4.68E-01	4.28E-01
Zr-97+D	5.33E-06	4.66E-05	5.87E-06	4.68E-06	4.38E-06	5.26E-06	4.77E-06
Nb-96	5.17E-09	9.81E-09	5.72E-09	4.56E-09	4.26E-09	5.06E-09	4.62E-09
Mo-99+D	2.31E-04	5.12E-03	2.55E-04	2.00E-04	1.87E-04	2.34E-04	2.02E-04
Ru-103+D	2.85E-02	4.19E-02	3.08E-02	2.48E-02	2.31E-02	2.91E-02	2.53E-02
Ru-105+D	1.52E-07	8.47E-07	1.67E-07	1.33E-07	1.24E-07	1.53E-07	1.35E-07
Ru-106+D	7.94E-03	5.90E-01	8.66E-03	6.96E-03	6.51E-03	7.98E-03	7.11E-03
Sb-129+D	1.76E-07	1.26E-06	1.95E-07	1.56E-07	1.46E-07	1.73E-07	1.59E-07
Te-131	7.85E-08	1.38E-06	8.66E-08	6.73E-08	6.25E-08	8.35E-08	6.70E-08
Te-131m+D	7.40E-06	2.06E-05	8.19E-06	6.50E-06	6.08E-06	7.33E-06	6.54E-06
Te-132+D	4.24E-03	1.32E-02	4.67E-03	3.69E-03	3.46E-03	4.21E-03	3.75E-03
Te-133m+D	6.67E-07	3.29E-06	7.40E-07	5.87E-07	5.51E-07	6.56E-07	5.94E-07
Te-134	4.48E-07	6.21E-07	4.92E-07	3.87E-07	3.59E-07	4.64E-07	3.90E-07
I-131 (ELE)	3.72E-01	9.77E-01	4.04E-01	3.20E-01	2.97E-01	3.94E-01	3.29E-01
I-132	1.56E-02	5.15E-02	1.72E-02	1.37E-02	1.28E-02	1.52E-02	1.39E-02
I-133	1.02E-03	7.23E-03	1.11E-03	8.90E-04	8.32E-04	1.02E-03	9.10E-04
I-134	2.08E-04	7.66E-04	2.31E-04	1.84E-04	1.72E-04	2.02E-04	1.86E-04
I-135	2.33E-04	7.95E-04	2.55E-04	2.07E-04	1.94E-04	2.22E-04	2.09E-04
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	3.11E-01	9.90E-01	3.43E-01	2.73E-01	2.55E-01	3.09E-01	2.78E-01
Cs-138	1.12E-06	8.36E-06	1.22E-06	9.97E-07	9.45E-07	1.07E-06	1.01E-06
Ba-139	3.45E-08	8.49E-06	3.81E-08	2.83E-08	2.59E-08	4.09E-08	2.70E-08
Ba-140+D	7.57E-02	3.20E-01	8.27E-02	6.70E-02	6.33E-02	7.32E-02	6.77E-02
Ba-141	1.12E-07	1.25E-06	1.23E-07	9.70E-08	9.06E-08	1.15E-07	9.84E-08
Ba-142	2.18E-08	1.29E-07	2.42E-08	1.91E-08	1.79E-08	2.16E-08	1.92E-08
La-141	3.67E-08	9.79E-06	4.04E-08	3.27E-08	3.09E-08	3.48E-08	3.29E-08
La-142	2.08E-06	1.02E-05	2.23E-06	1.85E-06	1.77E-06	1.96E-06	1.89E-06
Ce-141	7.60E-03	2.38E-02	8.49E-03	6.08E-03	5.52E-03	9.53E-03	5.46E-03
Ce-143	3.54E-05	4.86E-04	3.87E-05	2.97E-05	2.74E-05	3.76E-05	2.92E-05
Ce-144+D	2.28E-02	5.80E+00	2.51E-02	1.92E-02	1.79E-02	2.47E-02	1.81E-02
Pr-143	2.92E-10	7.32E-02	3.22E-10	2.57E-10	2.41E-10	2.87E-10	2.62E-10
Pr-144	5.67E-07	2.49E-04	6.13E-07	5.05E-07	4.78E-07	5.41E-07	5.12E-07
TOTALS	1.33E+00	1.18E+01	1.46E+00	1.16E+00	1.08E+00	1.34E+00	1.18E+00

Table C-31. (continued).

Report Date: 27-Jul-91
Adult receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	1.99E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	5.24E-39	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	7.77E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	3.73E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.59E-01	2.01E-01
Sr-91	1.17E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.34E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	8.55E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	3.94E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	9.58E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.70E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-97	8.64E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	1.40E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	4.95E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	4.58E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-105	3.99E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	4.44E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-129	4.31E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	1.51E-27	0.00E+00	2.49E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	3.66E-07	0.00E+00	3.41E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	7.70E-05	0.00E+00	2.29E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	2.98E-17	0.00E+00	6.27E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	4.03E-20	0.00E+00	6.34E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	9.12E-01	0.00E+00	3.10E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	2.56E-08	0.00E+00	6.28E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	1.40E-04	0.00E+00	4.76E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	8.07E-16	0.00E+00	9.76E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.83E-07	0.00E+00	9.48E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	5.63E-02	0.00E+00	5.40E-02	5.40E-02	0.00E+00	5.40E-02	5.40E-02
Cs-138	2.65E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	1.93E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.21E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	4.41E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	7.02E-52	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	1.81E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	1.40E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	6.53E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	2.63E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	4.63E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	6.49E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.58E-33	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	1.11E+00	0.00E+00	3.10E+01	5.40E-02	0.00E+00	5.13E-01	2.55E-01

Table C-32. Dose evaluation for the SL-1 Accident (infant)

Report Date: 27-Jul-91
File Name: ACC8B.WK1

Episodic Release

Episode Description..... SL-1 Accident
Date..... 1/3/61
Receptor Location..... Atomic City
Receptor Designation..... Infant
Release Location..... SL-1

Input Parameters

8.00E-11 : Episodic Mesoscale Dispersion Factor (hr/m**3)
1.60E+01 : Exposure Period (hr)
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 27-Jul-91

1/3/61 Episodic Release
SL-1 Accident

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	7.11E-02	1.82E-01	7.67E-02	6.31E-02	5.98E-02	6.84E-02	6.42E-02
Inhalation	7.79E-01	7.85E-03	8.69E+00	4.00E+00	4.56E-02	8.47E-02	7.47E-02
Grnd Surface	1.33E+00	1.18E+01	1.46E+00	1.16E+00	1.08E+00	1.34E+00	1.18E+00
Ingestion	5.28E+00	5.71E-02	1.73E+02	8.02E-02	7.27E-02	5.02E-01	3.93E-01
TOTALS	7.45E+00	1.21E+01	1.83E+02	5.30E+00	1.26E+00	1.99E+00	1.71E+00

Table C-32. (continued).

Report Date: 27-Jul-91
Infant receptor

1/3/61 Episodic Release
SL-1 Accident

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Br-84	5.30E-01	1.90E-02	9.50E-14	1.52E-12	1.80E-03
Kr-85m	4.48E+00	3.55E+01	1.78E-10	2.84E-09	0.00E+00
Kr-87	1.27E+00	2.52E+02	1.26E-09	2.02E-08	0.00E+00
Kr-88	2.84E+00	2.14E+02	1.07E-09	1.71E-08	0.00E+00
Rb-89	2.53E-01	5.46E-02	2.73E-13	4.37E-12	1.80E-03
Sr-89	1.21E+03	3.25E+00	1.63E-11	2.60E-10	1.80E-03
Sr-90	2.50E+05	1.00E-01	5.00E-13	8.00E-12	1.80E-03
Sr-91	9.50E+00	4.13E-03	2.07E-14	3.30E-13	1.80E-03
Sr-92	2.71E+00	1.21E-02	6.05E-14	9.68E-13	1.80E-03
Y-91	1.40E+03	1.95E+01	9.75E-11	1.56E-09	1.80E-03
Y-92	3.54E+00	1.19E-02	5.95E-14	9.52E-13	1.80E-03
Y-93	1.02E+01	2.06E-02	1.03E-13	1.65E-12	1.80E-03
Zr-95	1.54E+03	2.13E+01	1.07E-10	1.70E-09	1.80E-03
Zr-97	1.69E+01	3.28E-02	1.64E-13	2.62E-12	1.80E-03
Nb-96	2.34E+01	1.51E-05	7.55E-17	1.21E-15	1.80E-03
Mo-99	6.60E+01	3.66E+00	1.83E-11	2.93E-10	1.80E-03
Ru-103	9.44E+02	1.01E+01	5.05E-11	8.08E-10	1.80E-03
Ru-105	4.44E+00	7.06E-03	3.53E-14	5.65E-13	1.80E-03
Ru-106	8.84E+03	7.17E-01	3.59E-12	5.74E-11	1.80E-03
Sb-129	4.32E+00	4.67E-03	2.34E-14	3.74E-13	1.80E-03
Te-131	4.20E-01	7.18E-02	3.59E-13	5.74E-12	1.80E-03
Te-131m	3.00E+01	2.74E-02	1.37E-13	2.19E-12	1.80E-03
Te-132	7.80E+01	3.49E+00	1.75E-11	2.79E-10	1.80E-03
Te-133m	9.20E-01	5.18E-02	2.59E-13	4.14E-12	1.80E-03
Te-134	7.00E-01	1.18E-01	5.90E-13	9.44E-12	1.80E-03
I-131 (ELE)	1.93E+02	8.00E+01	4.00E-10	6.40E-09	1.80E-02
I-132	2.30E+00	4.95E+01	2.48E-10	3.96E-09	1.80E-02
I-133	2.08E+01	1.33E+00	6.65E-12	1.06E-10	1.80E-02
I-134	8.77E-01	1.54E+00	7.70E-12	1.23E-10	1.80E-02
I-135	6.61E+00	4.07E-01	2.04E-12	3.26E-11	1.80E-02
Xe-129m	2.13E+02	7.65E-05	3.83E-16	6.12E-15	0.00E+00
Xe-135	9.08E+00	2.09E+01	1.05E-10	1.67E-09	0.00E+00
Xe-135m	2.56E-01	3.90E+01	1.95E-10	3.12E-09	0.00E+00
Xe-138	2.36E-01	3.01E+02	1.51E-09	2.41E-08	0.00E+00
Cs-137	2.65E+05	5.00E-01	2.50E-12	4.00E-11	1.80E-03
Cs-138	5.37E-01	1.63E-01	8.15E-13	1.30E-11	1.80E-03
Ba-139	1.38E+00	1.08E-01	5.40E-13	8.64E-12	1.80E-03
Ba-140	3.07E+02	1.56E+01	7.80E-11	1.25E-09	1.80E-03
Ba-141	3.00E-01	6.98E-02	3.49E-13	5.58E-12	1.80E-03
Ba-142	1.80E-01	2.30E-02	1.15E-13	1.84E-12	1.80E-03
La-141	3.93E+00	4.13E-02	2.07E-13	3.30E-12	1.80E-03
La-142	1.59E+00	8.99E-02	4.50E-13	7.19E-12	1.80E-03
Ce-141	7.80E+02	1.86E+01	9.30E-11	1.49E-09	1.80E-03
Ce-143	3.30E+01	6.30E-01	3.15E-12	5.04E-11	1.80E-03
Ce-144	6.82E+03	1.10E+01	5.50E-11	8.80E-10	1.80E-03
Pr-143	3.25E+02	1.66E+01	8.30E-11	1.33E-09	1.80E-03
Pr-144	2.90E-01	1.10E+01	5.50E-11	8.80E-10	1.80E-03
TOTAL		1.13E+03			

Table C-32. (continued).

Report Date: 27-Jul-91
Infant Receptor

1/3/61 Episodic Release
SL-1 Accident

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	1.88E-06	4.97E-06	1.99E-06	1.80E-06	1.58E-06	1.75E-06	1.70E-06
Kr-85m	2.65E-04	8.48E-04	2.92E-04	2.18E-04	1.99E-04	3.21E-04	2.09E-04
Kr-87	1.03E-02	3.79E-02	1.10E-02	9.08E-03	8.62E-03	9.86E-03	9.28E-03
Kr-88	2.99E-02	8.16E-02	3.21E-02	2.67E-02	2.54E-02	2.82E-02	2.72E-02
Rb-89	5.86E-06	1.26E-05	6.35E-06	5.34E-06	4.90E-06	5.53E-06	5.27E-06
Sr-89	3.98E-03	1.39E-01	3.58E-06	1.28E-02	2.68E-06	3.10E-06	2.89E-06
Sr-90	4.87E-02	1.12E+00	4.12E-03	4.18E-03	4.12E-03	3.75E-01	2.66E-01
Sr-91	5.11E-07	2.35E-06	4.83E-07	7.93E-07	3.60E-07	4.30E-07	3.91E-07
Sr-92	1.12E-06	1.59E-06	1.10E-06	1.52E-06	8.38E-07	9.47E-07	8.96E-07
Y-91	1.37E-02	1.00E+00	6.05E-04	9.19E-02	4.56E-04	5.18E-04	4.88E-04
Y-92	2.98E-07	5.49E-06	2.26E-07	8.43E-07	1.69E-07	1.97E-07	1.83E-07
Y-93	7.46E-07	1.80E-05	2.15E-07	2.60E-06	1.64E-07	1.95E-07	1.77E-07
Zr-95	5.44E-01	6.65E-01	5.41E-01	7.42E-01	4.11E-01	5.27E-01	4.48E-01
Zr-97	9.44E-06	5.26E-05	8.58E-06	7.46E-06	6.40E-06	7.67E-06	6.96E-06
Nb-96	7.42E-09	1.24E-08	7.67E-09	7.24E-09	5.72E-09	6.78E-09	6.19E-09
Mo-99	4.37E-04	5.25E-03	2.85E-04	9.65E-04	2.09E-04	2.61E-04	2.25E-04
Ru-103	4.15E-02	4.31E-02	3.25E-02	9.16E-02	2.54E-02	3.09E-02	2.70E-02
Ru-105	4.50E-07	1.38E-06	4.55E-07	5.49E-07	3.37E-07	4.16E-07	3.68E-07
Ru-106	4.62E-02	5.92E-01	1.06E-02	2.71E-01	8.56E-03	1.03E-02	1.01E-02
Sb-129	5.43E-07	1.99E-06	5.62E-07	6.45E-07	4.21E-07	4.97E-07	4.57E-07
Te-131	1.58E-06	6.72E-06	3.55E-06	1.56E-06	1.20E-06	1.58E-06	1.28E-06
Te-131m	1.13E-05	2.37E-05	5.22E-05	1.10E-05	7.71E-06	9.29E-06	8.30E-06
Te-132	5.01E-03	1.39E-02	1.54E-02	4.34E-03	3.81E-03	4.63E-03	4.13E-03
Te-133m	6.83E-06	1.42E-05	1.35E-05	6.30E-06	5.43E-06	6.42E-06	5.85E-06
Te-134	5.38E-06	7.53E-06	8.73E-06	4.85E-06	4.21E-06	5.39E-06	4.58E-06
I-131 (ELE)	5.76E+00	1.00E+00	1.82E+02	3.89E-01	3.21E-01	4.28E-01	3.65E-01
I-132	2.33E-02	6.05E-02	6.49E-02	2.49E-02	1.76E-02	2.07E-02	1.90E-02
I-133	1.23E-03	7.32E-03	6.97E-03	9.23E-04	8.63E-04	1.06E-03	9.43E-04
I-134	4.04E-04	1.09E-03	4.67E-04	3.65E-04	3.33E-04	3.90E-04	3.59E-04
I-135	2.70E-04	8.46E-04	4.51E-04	2.43E-04	2.20E-04	2.51E-04	2.37E-04
Xe-129m	8.16E-11	6.71E-10	8.37E-11	5.09E-11	4.37E-11	8.02E-11	3.61E-11
Xe-135	2.38E-04	6.94E-04	2.59E-04	2.00E-04	1.85E-04	2.67E-04	2.04E-04
Xe-135m	7.65E-04	1.27E-03	8.29E-04	6.65E-04	6.19E-04	7.75E-04	6.79E-04
Xe-138	1.72E-02	3.54E-02	1.85E-02	1.53E-02	1.45E-02	1.66E-02	1.55E-02
Cs-137	3.47E-01	1.02E+00	3.76E-01	3.05E-01	2.88E-01	3.46E-01	3.26E-01
Cs-138	2.02E-05	4.79E-05	2.19E-05	1.90E-05	1.69E-05	1.91E-05	1.81E-05
Ba-139	4.31E-07	1.56E-05	2.34E-07	1.45E-06	1.61E-07	2.51E-07	1.68E-07
Ba-140	7.97E-02	3.24E-01	8.51E-02	7.01E-02	6.51E-02	7.71E-02	7.06E-02
Ba-141	3.10E-06	9.27E-06	3.32E-06	3.03E-06	2.47E-06	3.11E-06	2.68E-06
Ba-142	1.01E-06	2.12E-06	1.11E-06	9.37E-07	8.25E-07	9.89E-07	8.88E-07
La-141	4.03E-07	1.27E-05	1.33E-07	1.36E-06	5.10E-07	1.14E-07	1.08E-07
La-142	1.52E-05	3.24E-05	1.59E-05	1.46E-05	1.27E-05	1.40E-05	1.35E-05
Ce-141	9.89E-03	2.40E-02	8.56E-03	2.10E-02	5.56E-03	9.61E-03	5.51E-03
Ce-143	6.85E-05	5.13E-04	4.70E-05	1.48E-04	3.33E-05	4.56E-05	3.55E-05
Ce-144	4.66E-01	5.80E+00	2.75E-02	3.21E+00	5.71E-02	1.05E-01	1.14E-01
Pr-143	1.68E-03	7.35E-02	3.30E-10	1.01E-02	2.46E-10	2.94E-10	2.68E-10
Pr-144	2.44E-05	1.24E-03	2.00E-05	6.52E-05	1.56E-05	1.76E-05	1.68E-05

Table C-32. (continued).

Report Date: 27-Jul-91
Infant receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	1.77E-06	4.05E-06	1.89E-06	1.57E-06	1.50E-06	1.66E-06	1.61E-06
Kr-85m	2.65E-04	8.48E-04	2.92E-04	2.18E-04	1.99E-04	3.21E-04	2.09E-04
Kr-87	1.03E-02	3.79E-02	1.10E-02	9.08E-03	8.62E-03	9.86E-03	9.28E-03
Kr-88+D	2.99E-02	8.16E-02	3.21E-02	2.67E-02	2.54E-02	2.82E-02	2.72E-02
Rb-89	5.68E-06	1.15E-05	6.17E-06	5.03E-06	4.77E-06	5.38E-06	5.13E-06
Sr-89	2.13E-08	1.30E-04	2.37E-08	1.88E-08	1.76E-08	2.06E-08	1.91E-08
Sr-90+D	0.00E+00	7.74E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	1.99E-07	4.58E-07	2.19E-07	1.75E-07	1.64E-07	1.95E-07	1.78E-07
Sr-92	7.65E-07	1.15E-06	8.43E-07	6.80E-07	6.42E-07	7.26E-07	6.86E-07
Y-91	3.33E-06	8.13E-04	3.70E-06	2.95E-06	2.79E-06	3.17E-06	2.99E-06
Y-92	1.43E-07	1.47E-06	1.58E-07	1.26E-07	1.18E-07	1.38E-07	1.28E-07
Y-93	9.06E-08	1.90E-06	9.86E-08	7.98E-08	7.53E-08	8.89E-08	8.12E-08
Zr-95+D	2.43E-03	3.39E-03	2.70E-03	2.14E-03	2.01E-03	2.38E-03	2.18E-03
Zr-97+D	2.46E-06	5.99E-06	2.70E-06	2.16E-06	2.02E-06	2.42E-06	2.20E-06
Nb-96	1.76E-09	2.59E-09	1.96E-09	1.56E-09	1.46E-09	1.72E-09	1.57E-09
Mo-99+D	2.67E-05	1.34E-04	2.95E-05	2.33E-05	2.17E-05	2.70E-05	2.36E-05
Ru-103+D	2.23E-04	3.13E-04	2.41E-04	1.94E-04	1.81E-04	2.28E-04	1.99E-04
Ru-105+D	2.61E-07	5.33E-07	2.88E-07	2.28E-07	2.13E-07	2.62E-07	2.33E-07
Ru-106+D	6.87E-06	8.37E-05	7.52E-06	6.03E-06	5.63E-06	6.87E-06	6.15E-06
Sb-129+D	3.33E-07	7.34E-07	3.67E-07	2.94E-07	2.75E-07	3.24E-07	2.98E-07
Te-131	1.41E-06	5.34E-06	1.56E-06	1.22E-06	1.13E-06	1.49E-06	1.21E-06
Te-131m+D	1.98E-06	3.18E-06	2.19E-06	1.74E-06	1.63E-06	1.95E-06	1.76E-06
Te-132+D	4.25E-04	6.90E-04	4.67E-04	3.73E-04	3.48E-04	4.20E-04	3.76E-04
Te-133m+D	5.91E-06	1.09E-05	6.51E-06	5.20E-06	4.88E-06	5.77E-06	5.25E-06
Te-134	4.78E-06	6.91E-06	5.25E-06	4.13E-06	3.85E-06	4.93E-06	4.19E-06
I-131 (ELE)	1.39E-03	2.57E-03	1.51E-03	1.20E-03	1.11E-03	1.47E-03	1.23E-03
I-132	5.37E-03	8.80E-03	5.91E-03	4.74E-03	4.43E-03	5.24E-03	4.79E-03
I-133	3.72E-05	8.45E-05	4.06E-05	3.26E-05	3.04E-05	3.74E-05	3.32E-05
I-134	1.94E-04	3.24E-04	2.15E-04	1.71E-04	1.60E-04	1.88E-04	1.73E-04
I-135	3.10E-05	5.09E-05	3.40E-05	2.75E-05	2.60E-05	2.95E-05	2.78E-05
Xe-129m	8.16E-11	6.71E-10	8.37E-11	5.09E-11	4.37E-11	8.02E-11	3.61E-11
Xe-135	2.38E-04	6.94E-04	2.59E-04	2.00E-04	1.85E-04	2.67E-04	2.04E-04
Xe-135m	7.65E-04	1.27E-03	8.29E-04	6.65E-04	6.19E-04	7.75E-04	6.79E-04
Xe-138	1.72E-02	3.54E-02	1.85E-02	1.53E-02	1.45E-02	1.66E-02	1.55E-02
Cs-137+D	1.32E-05	2.38E-05	1.45E-05	1.16E-05	1.08E-05	1.31E-05	1.18E-05
Cs-138	1.89E-05	3.95E-05	2.07E-05	1.68E-05	1.59E-05	1.80E-05	1.71E-05
Ba-139	1.77E-07	7.09E-06	1.96E-07	1.47E-07	1.35E-07	2.10E-07	1.41E-07
Ba-140+D	2.15E-03	3.73E-03	2.34E-03	1.90E-03	1.79E-03	2.07E-03	1.93E-03
Ba-141	2.92E-06	8.02E-06	3.20E-06	2.55E-06	2.38E-06	2.99E-06	2.58E-06
Ba-142	9.80E-07	1.99E-06	1.09E-06	8.60E-07	8.08E-07	9.67E-07	8.68E-07
La-141	8.40E-08	2.90E-06	9.23E-08	7.46E-08	7.08E-08	7.95E-08	7.53E-08
La-142	1.29E-05	2.21E-05	1.37E-05	1.15E-05	1.09E-05	1.21E-05	1.16E-05
Ce-141	6.55E-05	2.02E-04	7.31E-05	5.28E-05	4.78E-05	8.23E-05	4.78E-05
Ce-143	7.58E-06	2.68E-05	8.27E-06	6.38E-06	5.92E-06	7.99E-06	6.38E-06
Ce-144+D	2.68E-05	1.01E-03	2.93E-05	2.29E-05	2.14E-05	2.85E-05	2.21E-05
Pr-143	6.99E-12	3.13E-04	7.74E-12	6.15E-12	5.75E-12	6.86E-12	6.25E-12
Pr-144	1.80E-05	9.90E-04	1.94E-05	1.60E-05	1.51E-05	1.71E-05	1.63E-05
TOTALS	7.11E-02	1.82E-01	7.67E-02	6.31E-02	5.98E-02	6.84E-02	6.42E-02

Table C-32. (continued).

Report Date: 27-Jul-91
Infant receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	2.09E-08	0.00E+00	0.00E+00	1.42E-07	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.56E-08	0.00E+00	0.00E+00	1.73E-07	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.52E-03	0.00E+00	0.00E+00	1.28E-02	0.00E+00	0.00E+00	0.00E+00
Sr-90	5.62E-04	4.69E-05	4.69E-05	1.08E-04	4.69E-05	4.46E-03	3.19E-03
Sr-91	7.32E-08	0.00E+00	0.00E+00	4.08E-07	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.18E-07	0.00E+00	0.00E+00	6.28E-07	0.00E+00	0.00E+00	0.00E+00
Y-91	1.09E-02	0.00E+00	0.00E+00	9.14E-02	0.00E+00	0.00E+00	0.00E+00
Y-92	9.34E-08	0.00E+00	0.00E+00	6.63E-07	0.00E+00	0.00E+00	0.00E+00
Y-93	5.48E-07	0.00E+00	0.00E+00	2.43E-06	0.00E+00	0.00E+00	0.00E+00
Zr-95	4.99E-02	3.80E-03	7.50E-03	3.18E-01	1.30E-02	2.50E-02	9.79E-03
Zr-97	1.66E-06	0.00E+00	0.00E+00	6.23E-07	0.00E+00	0.00E+00	0.00E+00
Nb-96	3.82E-10	0.00E+00	0.00E+00	1.13E-09	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.67E-04	0.00E+00	0.00E+00	7.42E-04	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.04E-02	6.15E-04	1.04E-03	6.63E-02	1.66E-03	1.14E-03	9.47E-04
Ru-105	3.67E-08	0.00E+00	0.00E+00	1.88E-07	0.00E+00	0.00E+00	0.00E+00
Ru-106	3.30E-02	5.04E-04	5.71E-04	2.62E-01	6.72E-04	6.72E-04	7.99E-04
Sb-129	3.37E-08	0.00E+00	0.00E+00	1.95E-07	0.00E+00	0.00E+00	0.00E+00
Te-131	9.00E-08	0.00E+00	1.91E-06	2.73E-07	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.91E-06	0.00E+00	4.16E-05	2.81E-06	0.00E+00	0.00E+00	0.00E+00
Te-132	3.40E-04	0.00E+00	1.02E-02	2.78E-04	0.00E+00	0.00E+00	0.00E+00
Te-133m	2.49E-07	0.00E+00	6.30E-06	5.12E-07	0.00E+00	0.00E+00	0.00E+00
Te-134	1.49E-07	0.00E+00	2.99E-06	3.29E-07	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	2.58E-01	1.12E-03	8.62E+00	2.62E-02	1.24E-03	1.65E-03	1.69E-03
I-132	2.32E-03	1.83E-04	4.17E-02	6.52E-03	3.25E-04	2.78E-04	2.78E-04
I-133	9.10E-05	0.00E+00	3.03E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	2.15E-06	0.00E+00	2.15E-05	1.01E-05	0.00E+00	0.00E+00	0.00E+00
I-135	5.67E-06	0.00E+00	1.60E-04	8.25E-06	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	3.05E-04	2.32E-04	2.81E-04	4.69E-04	2.81E-04	3.05E-04	3.98E-04
Cs-138	1.82E-07	0.00E+00	0.00E+00	1.22E-06	0.00E+00	0.00E+00	0.00E+00
Ba-139	2.19E-07	0.00E+00	0.00E+00	1.27E-06	0.00E+00	0.00E+00	0.00E+00
Ba-140	7.11E-04	0.00E+00	0.00E+00	1.24E-03	0.00E+00	1.76E-03	9.48E-04
Ba-141	6.54E-08	0.00E+00	0.00E+00	3.89E-07	0.00E+00	0.00E+00	0.00E+00
Ba-142	1.05E-08	0.00E+00	0.00E+00	5.83E-08	0.00E+00	0.00E+00	0.00E+00
La-141	2.82E-07	0.00E+00	0.00E+00	1.26E-06	4.08E-07	0.00E+00	0.00E+00
La-142	2.50E-07	0.00E+00	0.00E+00	1.25E-06	0.00E+00	0.00E+00	0.00E+00
Ce-141	2.00E-03	0.00E+00	0.00E+00	1.48E-02	0.00E+00	0.00E+00	0.00E+00
Ce-143	2.55E-05	0.00E+00	0.00E+00	1.12E-04	0.00E+00	0.00E+00	0.00E+00
Ce-144	4.07E-01	1.34E-03	1.60E-03	3.19E+00	2.84E-02	4.95E-02	5.67E-02
Pr-143	1.53E-03	0.00E+00	0.00E+00	1.01E-02	0.00E+00	0.00E+00	0.00E+00
Pr-144	5.85E-06	0.00E+00	0.00E+00	4.88E-05	0.00E+00	0.00E+00	0.00E+00
TOTALS	7.79E-01	7.85E-03	8.69E+00	4.00E+00	4.56E-02	8.47E-02	7.47E-02

Table C-32. (continued).

Report Date: 27-Jul-91
Infant receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	9.56E-08	9.20E-07	1.03E-07	8.48E-08	8.06E-08	9.02E-08	8.66E-08
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.57E-07	1.09E-06	1.72E-07	1.40E-07	1.32E-07	1.49E-07	1.42E-07
Sr-89	3.20E-06	1.39E-01	3.55E-06	2.82E-06	2.66E-06	3.08E-06	2.87E-06
Sr-90+D	0.00E+00	1.12E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	2.39E-07	1.89E-06	2.64E-07	2.10E-07	1.97E-07	2.35E-07	2.14E-07
Sr-92	2.33E-07	4.46E-07	2.58E-07	2.07E-07	1.96E-07	2.21E-07	2.09E-07
Y-91	5.41E-04	1.00E+00	6.02E-04	4.81E-04	4.53E-04	5.15E-04	4.85E-04
Y-92	6.09E-08	4.02E-06	6.74E-08	5.38E-08	5.05E-08	5.88E-08	5.46E-08
Y-93	1.07E-07	1.61E-05	1.17E-07	9.43E-08	8.88E-08	1.06E-07	9.58E-08
Zr-95+D	4.78E-01	6.56E-01	5.30E-01	4.21E-01	3.94E-01	4.68E-01	4.28E-01
Zr-97+D	5.33E-06	4.66E-05	5.87E-06	4.68E-06	4.38E-06	5.26E-06	4.77E-06
Nb-96	5.17E-09	9.81E-09	5.72E-09	4.56E-09	4.26E-09	5.06E-09	4.62E-09
Mo-99+D	2.31E-04	5.12E-03	2.55E-04	2.00E-04	1.87E-04	2.34E-04	2.02E-04
Ru-103+D	2.85E-02	4.19E-02	3.08E-02	2.48E-02	2.31E-02	2.91E-02	2.53E-02
Ru-105+D	1.52E-07	8.47E-07	1.67E-07	1.33E-07	1.24E-07	1.53E-07	1.35E-07
Ru-106+D	7.94E-03	5.90E-01	8.66E-03	6.96E-03	6.51E-03	7.98E-03	7.11E-03
Sb-129+D	1.76E-07	1.26E-06	1.95E-07	1.56E-07	1.46E-07	1.73E-07	1.59E-07
Te-131	7.85E-08	1.38E-06	8.66E-08	6.73E-08	6.25E-08	8.35E-08	6.70E-08
Te-131m+D	7.40E-06	2.06E-05	8.19E-06	6.50E-06	6.08E-06	7.33E-06	6.54E-06
Te-132+D	4.24E-03	1.32E-02	4.67E-03	3.69E-03	3.46E-03	4.21E-03	3.75E-03
Te-133m+D	6.67E-07	3.29E-06	7.40E-07	5.87E-07	5.51E-07	6.56E-07	5.94E-07
Te-134	4.48E-07	6.21E-07	4.92E-07	3.87E-07	3.59E-07	4.64E-07	3.90E-07
I-131 (ELE)	3.72E-01	9.77E-01	4.04E-01	3.20E-01	2.97E-01	3.94E-01	3.29E-01
I-132	1.56E-02	5.15E-02	1.72E-02	1.37E-02	1.28E-02	1.52E-02	1.39E-02
I-133	1.02E-03	7.23E-03	1.11E-03	8.90E-04	8.32E-04	1.02E-03	9.10E-04
I-134	2.08E-04	7.66E-04	2.31E-04	1.84E-04	1.72E-04	2.02E-04	1.86E-04
I-135	2.33E-04	7.95E-04	2.55E-04	2.07E-04	1.94E-04	2.22E-04	2.09E-04
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	3.11E-01	9.90E-01	3.43E-01	2.73E-01	2.55E-01	3.09E-01	2.78E-01
Cs-138	1.12E-06	8.36E-06	1.22E-06	9.97E-07	9.45E-07	1.07E-06	1.01E-06
Ba-139	3.45E-08	8.49E-06	3.81E-08	2.83E-08	2.59E-08	4.09E-08	2.70E-08
Ba-140+D	7.57E-02	3.20E-01	8.27E-02	6.70E-02	6.33E-02	7.32E-02	6.77E-02
Ba-141	1.12E-07	1.25E-06	1.23E-07	9.70E-08	9.06E-08	1.15E-07	9.84E-08
Ba-142	2.18E-08	1.29E-07	2.42E-08	1.91E-08	1.79E-08	2.16E-08	1.92E-08
La-141	3.67E-08	9.79E-06	4.04E-08	3.27E-08	3.09E-08	3.48E-08	3.29E-08
La-142	2.08E-06	1.02E-05	2.23E-06	1.85E-06	1.77E-06	1.96E-06	1.89E-06
Ce-141	7.60E-03	2.38E-02	8.49E-03	6.08E-03	5.52E-03	9.53E-03	5.46E-03
Ce-143	3.54E-05	4.86E-04	3.87E-05	2.97E-05	2.74E-05	3.76E-05	2.92E-05
Ce-144+D	2.28E-02	5.80E+00	2.51E-02	1.92E-02	1.79E-02	2.47E-02	1.81E-02
Pr-143	2.92E-10	7.32E-02	3.22E-10	2.57E-10	2.41E-10	2.87E-10	2.62E-10
Pr-144	5.67E-07	2.49E-04	6.13E-07	5.05E-07	4.78E-07	5.41E-07	5.12E-07
TOTALS	1.33E+00	1.18E+01	1.46E+00	1.16E+00	1.08E+00	1.34E+00	1.18E+00

Table C-32. (continued).

Report Date: 27-Jul-91
Infant receptor

1/3/61 Episodic Release
SL-1 Accident

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Br-84	3.11E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	8.19E-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	2.46E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	4.81E-02	4.07E-03	4.07E-03	4.07E-03	4.07E-03	3.70E-01	2.63E-01
Sr-91	9.78E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.21E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.28E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	6.18E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	1.60E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.35E-02	8.90E-04	5.14E-04	8.90E-04	1.48E-03	3.10E-02	8.50E-03
Zr-97	1.62E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	1.08E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.30E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	2.31E-03	2.94E-04	3.29E-04	3.29E-04	4.49E-04	4.19E-04	4.79E-04
Ru-105	6.24E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	5.31E-03	1.31E-03	1.37E-03	1.37E-03	1.37E-03	1.61E-03	2.15E-03
Sb-129	7.00E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	2.36E-29	0.00E+00	3.89E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.70E-08	0.00E+00	1.59E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	4.67E-06	0.00E+00	1.39E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	4.65E-19	0.00E+00	9.80E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	6.30E-22	0.00E+00	9.90E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	5.13E+00	2.19E-02	1.73E+02	4.15E-02	2.14E-02	3.08E-02	3.26E-02
I-132	5.26E-09	4.39E-10	1.20E-07	8.77E-10	7.02E-10	6.14E-10	7.02E-10
I-133	8.20E-05	0.00E+00	2.79E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.26E-17	0.00E+00	1.53E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	5.67E-08	0.00E+00	1.90E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	3.48E-02	2.79E-02	3.31E-02	3.13E-02	3.31E-02	3.66E-02	4.70E-02
Cs-138	4.15E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.02E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.21E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	6.89E-36	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	1.10E-53	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	2.84E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	2.19E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	2.16E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	4.99E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	3.59E-02	7.18E-04	7.18E-04	7.18E-04	1.08E-02	3.09E-02	3.95E-02
Pr-143	1.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	2.47E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	5.28E+00	5.71E-02	1.73E+02	8.02E-02	7.27E-02	5.02E-01	3.93E-01

Table C-33. Dose evaluation for IET #23 (FEET #2) (adult)

Report Date: 24-Jun-91
File Name: ACC8A.WK1

Episodic Release

Episode Description..... IET #23 (FEET #2)
Date..... 9/7-10/14/60
Receptor Location..... Montview
Receptor Designation..... Adult
Release Location..... IET

Input Parameters

3.37E-12 : Episodic Mesoscale Dispersion Factor (hr/m**3)
3.10E+02 : Exposure Period (hr)
9.59E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 24-Jun-91

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	4.95E-03	1.08E-02	5.41E-03	4.40E-03	4.17E-03	4.73E-03	4.46E-03
Inhalation	1.14E-03	0.00E+00	3.43E-02	9.54E-04	6.90E-07	3.28E-05	4.84E-06
Grnd Surface	3.23E-03	1.38E-02	3.54E-03	2.84E-03	2.65E-03	3.23E-03	2.89E-03
Ingestion	2.29E-02	0.00E+00	7.78E-01	2.20E-06	0.00E+00	6.07E-05	2.56E-05
TOTALS	3.22E-02	2.45E-02	8.21E-01	8.19E-03	6.82E-03	8.05E-03	7.38E-03

Table C-33. (continued).

Report Date: 24-Jun-91
Adult receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Ar-41	1.83E+00	7.24E+02	7.86E-12	2.44E-09	0.00E+00
Br-84	5.30E-01	1.15E+01	1.25E-13	3.87E-11	1.80E-03
Kr-85m	4.48E+00	3.38E+01	3.67E-13	1.14E-10	0.00E+00
Kr-87	1.27E+00	1.42E+02	1.54E-12	4.79E-10	0.00E+00
Kr-88	2.84E+00	1.77E+02	1.92E-12	5.96E-10	0.00E+00
Rb-89	2.53E-01	3.41E+01	3.70E-13	1.15E-10	1.80E-03
Sr-89	1.21E+03	4.64E-02	5.04E-16	1.56E-13	1.80E-03
Sr-90	2.50E+05	2.73E-04	2.96E-18	9.20E-16	1.80E-03
Sr-91	9.50E+00	2.86E-01	3.11E-15	9.64E-13	1.80E-03
Sr-92	2.71E+00	3.30E-01	3.58E-15	1.11E-12	1.80E-03
Y-91	1.40E+03	2.41E-02	2.62E-16	8.12E-14	1.80E-03
Y-92	3.54E+00	1.82E-01	1.97E-15	6.13E-13	1.80E-03
Y-93	1.02E+01	1.73E-01	1.87E-15	5.82E-13	1.80E-03
Zr-95	1.54E+03	2.63E-02	2.85E-16	8.85E-14	1.80E-03
Zr-97	1.69E+01	1.21E-01	1.32E-15	4.08E-13	1.80E-03
Nb-96	2.34E+01	1.02E-05	1.11E-19	3.45E-17	1.80E-03
Mo-99	6.60E+01	6.40E-02	6.94E-16	2.16E-13	1.80E-03
Ru-103	9.44E+02	1.58E-02	1.72E-16	5.33E-14	1.80E-03
Ru-105	4.44E+00	4.32E-02	4.69E-16	1.46E-13	1.80E-03
Ru-106	8.84E+03	4.53E-04	4.92E-18	1.53E-15	1.80E-03
Sb-129	4.32E+00	2.85E-02	3.09E-16	9.60E-14	1.80E-03
Te-131	4.20E-01	1.58E-01	1.71E-15	5.32E-13	1.80E-03
Te-131m	3.00E+01	5.55E-03	6.02E-17	1.87E-14	1.80E-03
Te-132	7.80E+01	4.32E-02	4.69E-16	1.46E-13	1.80E-03
Te-133m	9.20E-01	1.59E-01	1.73E-15	5.36E-13	1.80E-03
Te-134	7.00E-01	3.26E-01	3.54E-15	1.10E-12	1.80E-03
I-131 (ELE)	1.93E+02	4.55E+00	4.94E-14	1.53E-11	1.80E-02
I-132	2.30E+00	5.56E+00	6.04E-14	1.88E-11	1.80E-02
I-133	2.08E+01	2.27E+01	2.47E-13	7.66E-11	1.80E-02
I-134	8.77E-01	6.24E+01	6.78E-13	2.10E-10	1.80E-02
I-135	6.61E+00	4.48E+01	4.87E-13	1.51E-10	1.80E-02
Xe-129m	2.13E+02	4.02E-08	4.36E-22	1.35E-19	0.00E+00
Xe-135	9.08E+00	7.89E+01	8.57E-13	2.66E-10	0.00E+00
Xe-135m	2.56E-01	1.54E+01	1.67E-13	5.20E-11	0.00E+00
Xe-138	2.36E-01	1.60E+02	1.74E-12	5.41E-10	0.00E+00
Cs-137	2.65E+05	4.83E-04	5.25E-18	1.63E-15	1.80E-03
Cs-138	5.37E-01	1.32E+02	1.44E-12	4.46E-10	1.80E-03
Ba-139	1.38E+00	3.80E+00	4.13E-14	1.28E-11	1.80E-03
Ba-140	3.07E+02	4.64E-02	5.04E-16	1.56E-13	1.80E-03
Ba-141	3.00E-01	2.61E-01	2.83E-15	8.78E-13	1.80E-03
Ba-142	1.80E-01	1.22E-01	1.32E-15	4.11E-13	1.80E-03
La-141	3.93E+00	2.74E-01	2.97E-15	9.23E-13	1.80E-03
La-142	1.59E+00	3.63E-01	3.94E-15	1.22E-12	1.80E-03
Ce-141	7.80E+02	3.01E-02	3.27E-16	1.02E-13	1.80E-03
Ce-143	3.30E+01	8.65E-02	9.39E-16	2.92E-13	1.80E-03
Ce-144	6.82E+03	7.41E-03	8.05E-17	2.50E-14	1.80E-03
Pr-143	3.25E+02	3.11E-02	3.38E-16	1.05E-13	1.80E-03
Pr-144	2.90E-01	7.40E-03	8.04E-17	2.49E-14	1.80E-03
U-234	2.14E+09	1.97E-05	2.14E-19	6.64E-17	1.80E-03
U-235	6.17E+12	6.26E-07	6.80E-21	2.11E-18	1.80E-03
U-238	3.91E+13	5.82E-09	6.32E-23	1.96E-20	1.80E-03
TOTAL		1.66E+03			

Table C-33. (continued).

Report Date: 24-Jun-91
Adult receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	1.84E-03	3.39E-03	2.04E-03	1.64E-03	1.55E-03	1.75E-03	1.65E-03
Br-84	5.06E-05	1.27E-04	5.07E-05	6.41E-05	4.02E-05	4.46E-05	4.32E-05
Kr-85m	1.06E-05	3.40E-05	1.17E-05	8.73E-06	7.99E-06	1.28E-05	8.38E-06
Kr-87	2.44E-04	9.01E-04	2.61E-04	2.16E-04	2.05E-04	2.34E-04	2.21E-04
Kr-88	1.04E-03	2.84E-03	1.12E-03	9.30E-04	8.85E-04	9.84E-04	9.47E-04
Rb-89	1.57E-04	3.31E-04	1.67E-04	1.63E-04	1.29E-04	1.45E-04	1.39E-04
Sr-89	1.02E-05	8.34E-05	2.15E-09	4.65E-05	1.61E-09	1.87E-09	1.74E-09
Sr-90	5.43E-06	1.29E-04	0.00E+00	9.70E-06	0.00E+00	5.27E-05	2.31E-05
Sr-91	2.61E-06	6.87E-06	1.41E-06	8.34E-06	1.05E-06	1.26E-06	1.14E-06
Sr-92	1.96E-06	1.83E-06	1.26E-06	5.38E-06	9.61E-07	1.09E-06	1.03E-06
Y-91	3.50E-06	5.23E-05	3.15E-08	2.88E-05	2.37E-08	2.70E-08	2.54E-08
Y-92	4.96E-07	3.53E-06	1.45E-07	2.70E-06	1.09E-07	1.27E-07	1.18E-07
Y-93	1.27E-06	6.34E-06	7.60E-08	5.25E-06	5.79E-08	6.88E-08	6.25E-08
Zr-95	2.74E-05	3.43E-05	2.77E-05	2.20E-05	2.06E-05	5.59E-05	2.64E-05
Zr-97	2.91E-06	8.19E-06	1.34E-06	6.94E-06	9.96E-07	1.19E-06	1.08E-06
Nb-96	3.04E-10	3.54E-10	2.19E-10	3.69E-10	1.63E-10	1.94E-10	1.77E-10
Mo-99	1.30E-06	3.87E-06	2.10E-07	3.47E-06	1.54E-07	1.92E-07	1.66E-07
Ru-103	2.60E-06	2.78E-06	2.05E-06	4.66E-06	1.53E-06	1.93E-06	1.68E-06
Ru-105	1.64E-07	3.56E-07	1.17E-07	3.86E-07	8.70E-08	1.07E-07	9.49E-08
Ru-106	9.74E-07	1.57E-05	2.31E-07	5.60E-06	1.73E-07	2.13E-07	1.89E-07
Sb-129	1.83E-07	5.12E-07	1.44E-07	4.19E-07	1.08E-07	1.28E-07	1.17E-07
Te-131	1.89E-07	6.22E-07	1.22E-06	2.72E-07	1.11E-07	1.46E-07	1.18E-07
Te-131m	2.10E-07	2.02E-07	2.53E-06	2.16E-07	6.57E-08	7.92E-08	7.08E-08
Te-132	3.91E-06	7.25E-06	4.67E-05	3.00E-06	1.99E-06	2.41E-06	2.15E-06
Te-133m	1.05E-06	1.84E-06	5.87E-06	1.15E-06	7.02E-07	8.31E-07	7.56E-07
Te-134	7.14E-07	8.77E-07	2.78E-06	7.58E-07	4.90E-07	6.28E-07	5.33E-07
I-131 (ELE)	2.32E-02	2.35E-03	7.60E-01	7.70E-04	7.14E-04	9.49E-04	7.92E-04
I-132	1.05E-04	2.86E-04	2.23E-04	1.05E-04	8.18E-05	9.69E-05	8.86E-05
I-133	2.16E-03	5.27E-03	4.83E-02	6.64E-04	6.21E-04	7.64E-04	6.79E-04
I-134	7.08E-04	1.86E-03	9.83E-04	7.11E-04	5.68E-04	6.67E-04	6.13E-04
I-135	1.40E-03	3.92E-03	6.27E-03	1.32E-03	1.02E-03	1.17E-03	1.10E-03
Xe-129m	1.80E-15	1.48E-14	1.85E-15	1.12E-15	9.66E-16	1.77E-15	7.99E-16
Xe-135	3.79E-05	1.10E-04	4.12E-05	3.18E-05	2.95E-05	4.24E-05	3.24E-05
Xe-135m	1.27E-05	2.12E-05	1.38E-05	1.11E-05	1.03E-05	1.29E-05	1.13E-05
Xe-138	3.86E-04	7.95E-04	4.16E-04	3.43E-04	3.25E-04	3.73E-04	3.48E-04
Cs-137	1.50E-05	4.03E-05	1.62E-05	1.34E-05	1.04E-05	1.48E-05	1.36E-05
Cs-138	7.22E-04	1.64E-03	7.49E-04	8.62E-04	5.77E-04	6.52E-04	6.20E-04
Ba-139	2.28E-06	2.31E-05	3.47E-07	1.17E-05	2.39E-07	3.72E-07	2.49E-07
Ba-140	1.18E-05	4.06E-05	1.07E-05	9.58E-06	8.17E-06	1.08E-05	9.45E-06
Ba-141	5.39E-07	1.46E-06	5.23E-07	7.86E-07	3.89E-07	4.88E-07	4.22E-07
Ba-142	2.38E-07	4.73E-07	2.48E-07	2.75E-07	1.84E-07	2.21E-07	1.98E-07
La-141	5.12E-07	3.54E-06	3.71E-08	2.15E-06	7.19E-07	3.19E-08	3.02E-08
La-142	2.80E-06	5.51E-06	2.71E-06	3.56E-06	2.16E-06	2.38E-06	2.30E-06
Ce-141	1.80E-06	1.64E-06	5.84E-07	6.55E-06	3.80E-07	6.56E-07	3.76E-07
Ce-143	1.30E-06	2.97E-06	2.72E-07	4.12E-06	1.93E-07	2.64E-07	2.06E-07
Ce-144	1.03E-05	1.65E-04	7.14E-07	7.00E-05	5.07E-07	7.03E-07	5.15E-07
Pr-143	1.25E-06	5.80E-06	2.61E-14	4.82E-06	1.94E-14	2.32E-14	2.11E-14
Pr-144	1.53E-09	3.51E-08	5.66E-10	8.84E-09	4.43E-10	4.99E-10	4.75E-10
U-234	8.64E-06	9.57E-09	3.47E-10	7.00E-05	1.40E-10	5.65E-06	3.72E-07
U-235	2.63E-07	1.38E-08	9.69E-09	2.03E-06	6.46E-09	1.73E-07	1.76E-08
U-238	2.37E-09	6.40E-09	1.74E-11	1.88E-08	1.17E-11	1.52E-09	1.13E-10

Table C-33. (continued).

Report Date: 24-Jun-91
Adult receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	1.84E-03	3.39E-03	2.04E-03	1.64E-03	1.55E-03	1.75E-03	1.65E-03
Br-84	4.50E-05	1.03E-04	4.81E-05	4.00E-05	3.81E-05	4.23E-05	4.10E-05
Kr-85m	1.06E-05	3.40E-05	1.17E-05	8.73E-06	7.99E-06	1.28E-05	8.38E-06
Kr-87	2.44E-04	9.01E-04	2.61E-04	2.16E-04	2.05E-04	2.34E-04	2.21E-04
Kr-88+D	1.04E-03	2.84E-03	1.12E-03	9.30E-04	8.85E-04	9.84E-04	9.47E-04
Rb-89	1.49E-04	3.03E-04	1.62E-04	1.32E-04	1.25E-04	1.41E-04	1.35E-04
Sr-89	1.28E-11	7.80E-08	1.43E-11	1.13E-11	1.06E-11	1.24E-11	1.15E-11
Sr-90+D	0.00E+00	8.90E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	5.80E-07	1.34E-06	6.41E-07	5.10E-07	4.77E-07	5.69E-07	5.18E-07
Sr-92	8.77E-07	1.32E-06	9.67E-07	7.80E-07	7.37E-07	8.33E-07	7.88E-07
Y-91	1.73E-10	4.23E-08	1.93E-10	1.54E-10	1.45E-10	1.65E-10	1.56E-10
Y-92	9.22E-08	9.43E-07	1.02E-07	8.11E-08	7.62E-08	8.87E-08	8.25E-08
Y-93	3.20E-08	6.70E-07	3.48E-08	2.82E-08	2.66E-08	3.14E-08	2.86E-08
Zr-95+D	1.26E-07	1.76E-07	1.40E-07	1.11E-07	1.04E-07	1.24E-07	1.13E-07
Zr-97+D	3.82E-07	9.33E-07	4.21E-07	3.36E-07	3.14E-07	3.77E-07	3.42E-07
Nb-96	5.03E-11	7.39E-11	5.58E-11	4.44E-11	4.17E-11	4.91E-11	4.48E-11
Mo-99+D	1.96E-08	9.85E-08	2.17E-08	1.72E-08	1.60E-08	1.99E-08	1.74E-08
Ru-103+D	1.47E-08	2.06E-08	1.59E-08	1.28E-08	1.19E-08	1.50E-08	1.31E-08
Ru-105+D	6.73E-08	1.37E-07	7.42E-08	5.89E-08	5.50E-08	6.76E-08	6.00E-08
Ru-106+D	1.83E-10	2.23E-09	2.00E-10	1.61E-10	1.50E-10	1.83E-10	1.64E-10
Sb-129+D	8.54E-08	1.88E-07	9.44E-08	7.55E-08	7.08E-08	8.32E-08	7.65E-08
Te-131	1.31E-07	4.94E-07	1.44E-07	1.13E-07	1.05E-07	1.38E-07	1.12E-07
Te-131m+D	1.69E-08	2.71E-08	1.87E-08	1.48E-08	1.39E-08	1.67E-08	1.50E-08
Te-132+D	2.21E-07	3.60E-07	2.44E-07	1.94E-07	1.81E-07	2.19E-07	1.96E-07
Te-133m+D	7.64E-07	1.41E-06	8.42E-07	6.72E-07	6.31E-07	7.46E-07	6.79E-07
Te-134	5.56E-07	8.04E-07	6.11E-07	4.81E-07	4.49E-07	5.74E-07	4.87E-07
I-131 (ELE)	3.34E-06	6.16E-06	3.62E-06	2.87E-06	2.66E-06	3.53E-06	2.96E-06
I-132	2.54E-05	4.17E-05	2.80E-05	2.24E-05	2.10E-05	2.48E-05	2.27E-05
I-133	2.68E-05	6.08E-05	2.92E-05	2.35E-05	2.19E-05	2.69E-05	2.39E-05
I-134	3.31E-04	5.54E-04	3.67E-04	2.93E-04	2.73E-04	3.21E-04	2.95E-04
I-135	1.44E-04	2.36E-04	1.58E-04	1.28E-04	1.21E-04	1.37E-04	1.29E-04
Xe-129m	1.80E-15	1.48E-14	1.85E-15	1.12E-15	9.66E-16	1.77E-15	7.99E-16
Xe-135	3.79E-05	1.10E-04	4.12E-05	3.18E-05	2.95E-05	4.24E-05	3.24E-05
Xe-135m	1.27E-05	2.12E-05	1.38E-05	1.11E-05	1.03E-05	1.29E-05	1.13E-05
Xe-138	3.86E-04	7.95E-04	4.16E-04	3.43E-04	3.25E-04	3.73E-04	3.48E-04
Cs-137+D	5.38E-10	9.71E-10	5.92E-10	4.71E-10	4.41E-10	5.34E-10	4.81E-10
Cs-138	6.46E-04	1.35E-03	7.07E-04	5.75E-04	5.45E-04	6.16E-04	5.85E-04
Ba-139	2.63E-07	1.05E-05	2.91E-07	2.18E-07	2.00E-07	3.11E-07	2.09E-07
Ba-140+D	2.69E-07	4.68E-07	2.93E-07	2.38E-07	2.25E-07	2.59E-07	2.42E-07
Ba-141	4.59E-07	1.26E-06	5.03E-07	4.00E-07	3.74E-07	4.70E-07	4.06E-07
Ba-142	2.19E-07	4.44E-07	2.43E-07	1.92E-07	1.80E-07	2.16E-07	1.94E-07
La-141	2.35E-08	8.10E-07	2.58E-08	2.08E-08	1.98E-08	2.22E-08	2.10E-08
La-142	2.19E-06	3.77E-06	2.33E-06	1.95E-06	1.86E-06	2.05E-06	1.98E-06
Ce-141	4.47E-09	1.38E-08	4.99E-09	3.60E-09	3.26E-09	5.62E-09	3.26E-09
Ce-143	4.39E-08	1.55E-07	4.79E-08	3.69E-08	3.42E-08	4.62E-08	3.69E-08
Ce-144+D	7.61E-10	2.87E-08	8.32E-10	6.50E-10	6.08E-10	8.08E-10	6.28E-10
Pr-143	5.52E-16	2.47E-08	6.10E-16	4.85E-16	4.54E-16	5.41E-16	4.93E-16
Pr-144	5.09E-10	2.81E-08	5.49E-10	4.52E-10	4.29E-10	4.83E-10	4.61E-10
U-234	5.79E-15	2.47E-14	4.60E-15	3.11E-15	2.62E-15	5.37E-15	2.09E-15
U-235+D	2.00E-13	3.04E-13	2.20E-13	1.62E-13	1.48E-13	2.43E-13	1.53E-13
U-238+D	2.21E-16	1.47E-14	2.50E-16	1.85E-16	1.71E-16	2.50E-16	1.66E-16
TOTALS	4.95E-03	1.08E-02	5.41E-03	4.40E-03	4.17E-03	4.73E-03	4.46E-03

Table C-33. (continued).

Report Date: 24-Jun-91
Adult receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	3.23E-06	0.00E+00	0.00E+00	2.19E-05	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.08E-06	0.00E+00	0.00E+00	2.75E-05	0.00E+00	0.00E+00	0.00E+00
Sr-89	5.55E-06	0.00E+00	0.00E+00	4.65E-05	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.15E-06	0.00E+00	0.00E+00	9.70E-06	0.00E+00	0.00E+00	0.00E+00
Sr-91	1.29E-06	0.00E+00	0.00E+00	7.21E-06	0.00E+00	0.00E+00	0.00E+00
Sr-92	8.20E-07	0.00E+00	0.00E+00	4.37E-06	0.00E+00	0.00E+00	0.00E+00
Y-91	3.43E-06	0.00E+00	0.00E+00	2.88E-05	0.00E+00	0.00E+00	0.00E+00
Y-92	3.64E-07	0.00E+00	0.00E+00	2.59E-06	0.00E+00	0.00E+00	0.00E+00
Y-93	1.17E-06	0.00E+00	0.00E+00	5.19E-06	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.61E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.14E-05	4.08E-06
Zr-97	1.57E-06	0.00E+00	0.00E+00	5.88E-06	0.00E+00	0.00E+00	0.00E+00
Nb-96	6.61E-11	0.00E+00	0.00E+00	1.95E-10	0.00E+00	0.00E+00	0.00E+00
Mo-99	7.44E-07	0.00E+00	0.00E+00	3.31E-06	0.00E+00	0.00E+00	0.00E+00
Ru-103	3.98E-07	0.00E+00	0.00E+00	3.01E-06	0.00E+00	0.00E+00	0.00E+00
Ru-105	5.72E-08	0.00E+00	0.00E+00	2.93E-07	0.00E+00	0.00E+00	0.00E+00
Ru-106	6.44E-07	0.00E+00	0.00E+00	5.42E-06	0.00E+00	0.00E+00	0.00E+00
Sb-129	5.25E-08	0.00E+00	0.00E+00	3.04E-07	0.00E+00	0.00E+00	0.00E+00
Te-131	5.05E-08	0.00E+00	1.07E-06	1.53E-07	0.00E+00	0.00E+00	0.00E+00
Te-131m	9.86E-08	0.00E+00	2.15E-06	1.45E-07	0.00E+00	0.00E+00	0.00E+00
Te-132	1.07E-06	0.00E+00	3.21E-05	8.79E-07	0.00E+00	0.00E+00	0.00E+00
Te-133m	1.95E-07	0.00E+00	4.93E-06	4.01E-07	0.00E+00	0.00E+00	0.00E+00
Te-134	1.05E-07	0.00E+00	2.11E-06	2.32E-07	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	4.71E-04	0.00E+00	1.62E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	5.93E-06	0.00E+00	1.13E-04	1.80E-05	0.00E+00	0.00E+00	0.00E+00
I-133	3.97E-04	0.00E+00	1.32E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	2.22E-05	0.00E+00	2.22E-04	1.05E-04	0.00E+00	0.00E+00	0.00E+00
I-135	1.59E-04	0.00E+00	4.49E-03	2.32E-04	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	5.00E-08	0.00E+00	4.53E-08	5.16E-08	0.00E+00	4.53E-08	4.84E-08
Cs-138	3.77E-05	0.00E+00	0.00E+00	2.53E-04	0.00E+00	0.00E+00	0.00E+00
Ba-139	1.97E-06	0.00E+00	0.00E+00	1.14E-05	0.00E+00	0.00E+00	0.00E+00
Ba-140	5.40E-07	0.00E+00	0.00E+00	9.46E-07	0.00E+00	1.34E-06	7.20E-07
Ba-141	6.23E-08	0.00E+00	0.00E+00	3.70E-07	0.00E+00	0.00E+00	0.00E+00
Ba-142	1.42E-08	0.00E+00	0.00E+00	7.88E-08	0.00E+00	0.00E+00	0.00E+00
La-141	4.78E-07	0.00E+00	0.00E+00	2.12E-06	6.90E-07	0.00E+00	0.00E+00
La-142	2.58E-07	0.00E+00	0.00E+00	1.29E-06	0.00E+00	0.00E+00	0.00E+00
Ce-141	8.28E-07	0.00E+00	0.00E+00	6.14E-06	0.00E+00	0.00E+00	0.00E+00
Ce-143	8.95E-07	0.00E+00	0.00E+00	3.91E-06	0.00E+00	0.00E+00	0.00E+00
Ce-144	8.38E-06	0.00E+00	0.00E+00	6.95E-05	0.00E+00	0.00E+00	0.00E+00
Pr-143	7.34E-07	0.00E+00	0.00E+00	4.82E-06	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.00E-09	0.00E+00	0.00E+00	8.37E-09	0.00E+00	0.00E+00	0.00E+00
U-234	8.28E-06	0.00E+00	0.00E+00	7.00E-05	0.00E+00	0.00E+00	0.00E+00
U-235	2.43E-07	0.00E+00	0.00E+00	2.02E-06	0.00E+00	0.00E+00	0.00E+00
U-238	2.26E-09	0.00E+00	0.00E+00	1.88E-08	0.00E+00	0.00E+00	0.00E+00
TOTALS	1.14E-03	0.00E+00	3.43E-02	9.54E-04	6.90E-07	3.28E-05	4.84E-06

Table C-33. (continued).

Report Date: 24-Jun-91
Adult receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	2.43E-06	2.34E-05	2.62E-06	2.16E-06	2.05E-06	2.30E-06	2.20E-06
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.13E-06	2.87E-05	4.52E-06	3.67E-06	3.47E-06	3.93E-06	3.73E-06
Sr-89	1.92E-09	8.33E-05	2.14E-09	1.70E-09	1.60E-09	1.85E-09	1.73E-09
Sr-90+D	0.00E+00	1.29E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	6.98E-07	5.53E-06	7.70E-07	6.14E-07	5.75E-07	6.87E-07	6.24E-07
Sr-92	2.67E-07	5.12E-07	2.96E-07	2.38E-07	2.25E-07	2.54E-07	2.40E-07
Y-91	2.82E-08	5.22E-05	3.13E-08	2.50E-08	2.36E-08	2.68E-08	2.53E-08
Y-92	3.92E-08	2.59E-06	4.34E-08	3.46E-08	3.25E-08	3.79E-08	3.51E-08
Y-93	3.78E-08	5.67E-06	4.12E-08	3.33E-08	3.13E-08	3.74E-08	3.38E-08
Zr-95+D	2.48E-05	3.41E-05	2.75E-05	2.19E-05	2.05E-05	2.43E-05	2.22E-05
Zr-97+D	8.29E-07	7.26E-06	9.14E-07	7.29E-07	6.82E-07	8.18E-07	7.42E-07
Nb-96	1.47E-10	2.80E-10	1.63E-10	1.30E-10	1.22E-10	1.44E-10	1.32E-10
Mo-99+D	1.70E-07	3.77E-06	1.88E-07	1.48E-07	1.38E-07	1.72E-07	1.49E-07
Ru-103+D	1.88E-06	2.76E-06	2.03E-06	1.63E-06	1.52E-06	1.92E-06	1.67E-06
Ru-105+D	3.93E-08	2.18E-07	4.31E-08	3.43E-08	3.20E-08	3.95E-08	3.49E-08
Ru-106+D	2.11E-07	1.57E-05	2.31E-07	1.85E-07	1.73E-07	2.12E-07	1.89E-07
Sb-129+D	4.53E-08	3.23E-07	5.01E-08	4.01E-08	3.75E-08	4.44E-08	4.10E-08
Te-131	7.27E-09	1.28E-07	8.02E-09	6.24E-09	5.79E-09	7.74E-09	6.20E-09
Te-131m+D	6.31E-08	1.75E-07	6.98E-08	5.54E-08	5.18E-08	6.26E-08	5.58E-08
Te-132+D	2.21E-06	6.89E-06	2.44E-06	1.92E-06	1.81E-06	2.19E-06	1.95E-06
Te-133m+D	8.62E-08	4.25E-07	9.56E-08	7.60E-08	7.12E-08	8.48E-08	7.68E-08
Te-134	5.21E-08	7.23E-08	5.73E-08	4.50E-08	4.18E-08	5.40E-08	4.54E-08
I-131 (ELE)	8.93E-04	2.34E-03	9.68E-04	7.67E-04	7.12E-04	9.46E-04	7.89E-04
I-132	7.37E-05	2.44E-04	8.14E-05	6.47E-05	6.08E-05	7.21E-05	6.60E-05
I-133	7.32E-04	5.20E-03	7.99E-04	6.41E-04	5.99E-04	7.37E-04	6.55E-04
I-134	3.55E-04	1.31E-03	3.94E-04	3.14E-04	2.95E-04	3.46E-04	3.18E-04
I-135	1.08E-03	3.69E-03	1.18E-03	9.61E-04	9.02E-04	1.03E-03	9.69E-04
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	1.27E-05	4.03E-05	1.40E-05	1.11E-05	1.04E-05	1.26E-05	1.13E-05
Cs-138	3.84E-05	2.86E-04	4.18E-05	3.41E-05	3.24E-05	3.66E-05	3.45E-05
Ba-139	5.11E-08	1.26E-05	5.65E-08	4.20E-08	3.85E-08	6.06E-08	4.01E-08
Ba-140+D	9.49E-06	4.01E-05	1.04E-05	8.40E-06	7.94E-06	9.18E-06	8.49E-06
Ba-141	1.76E-08	1.97E-07	1.93E-08	1.53E-08	1.42E-08	1.81E-08	1.55E-08
Ba-142	4.87E-09	2.88E-08	5.40E-09	4.27E-09	4.00E-09	4.82E-09	4.29E-09
La-141	1.03E-08	2.73E-06	1.13E-08	9.12E-09	8.63E-09	9.72E-09	9.20E-09
La-142	3.54E-07	1.74E-06	3.79E-07	3.15E-07	3.01E-07	3.34E-07	3.22E-07
Ce-141	5.19E-07	1.63E-06	5.79E-07	4.15E-07	3.77E-07	6.50E-07	3.73E-07
Ce-143	2.05E-07	2.81E-06	2.24E-07	1.72E-07	1.59E-07	2.18E-07	1.69E-07
Ce-144+D	6.48E-07	1.65E-04	7.13E-07	5.46E-07	5.07E-07	7.02E-07	5.14E-07
Pr-143	2.31E-14	5.78E-06	2.54E-14	2.03E-14	1.90E-14	2.26E-14	2.06E-14
Pr-144	1.61E-11	7.07E-09	1.74E-11	1.43E-11	1.35E-11	1.53E-11	1.45E-11
U-234	1.21E-09	9.57E-09	3.47E-10	2.62E-10	1.40E-10	4.44E-10	1.38E-10
U-235+D	9.00E-09	1.38E-08	9.69E-09	7.11E-09	6.46E-09	1.07E-08	6.66E-09
U-238+D	1.61E-11	6.40E-09	1.74E-11	1.28E-11	1.17E-11	1.78E-11	1.12E-11
TOTALS	3.23E-03	1.38E-02	3.54E-03	2.84E-03	2.65E-03	3.23E-03	2.89E-03

Table C-33. (continued).

Report Date: 24-Jun-91
Adult receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	5.07E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.38E-36	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	4.67E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	4.28E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.27E-05	2.31E-05
Sr-91	3.41E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	8.42E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	4.45E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	2.54E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	3.38E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	8.83E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-97	1.35E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	3.99E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	3.64E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	3.02E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-105	1.03E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.18E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-129	1.11E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	1.40E-27	0.00E+00	2.30E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	3.12E-08	0.00E+00	2.91E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	4.01E-07	0.00E+00	1.19E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	3.85E-17	0.00E+00	8.11E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	4.70E-20	0.00E+00	7.38E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	2.19E-02	0.00E+00	7.43E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	1.21E-09	0.00E+00	2.97E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	1.01E-03	0.00E+00	3.43E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.38E-14	0.00E+00	1.67E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	1.31E-05	0.00E+00	4.40E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	2.29E-06	0.00E+00	2.20E-06	2.20E-06	0.00E+00	2.20E-06	2.20E-06
Cs-138	9.08E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	2.87E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.52E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	6.93E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	1.57E-51	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	5.05E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	2.39E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	4.46E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	1.52E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.31E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	5.12E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	4.48E-37	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-234	3.58E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.65E-06	3.72E-07
U-235	1.09E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.62E-07	1.09E-08
U-238	9.36E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.51E-09	1.02E-10
TOTALS	2.29E-02	0.00E+00	7.78E-01	2.20E-06	0.00E+00	6.07E-05	2.56E-05

Table C-34. Dose evaluation for IET #23 (FEET #2) (infant)

Report Date: 24-Jun-91
File Name: ACC8B.WK1

Episodic Release

Episode Description..... IET #23 (FEET #2)
Date..... 9/7-10/14/60
Receptor Location..... Montview
Receptor Designation..... Infant
Release Location..... IET

Input Parameters

3.37E-12 : Episodic Mesoscale Dispersion Factor (hr/m**3)
3.10E+02 : Exposure Period (hr)
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 24-Jun-91

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	4.95E-03	1.08E-02	5.41E-03	4.40E-03	4.17E-03	4.73E-03	4.46E-03
Inhalation	7.54E-04	3.87E-06	2.38E-02	3.51E-04	6.24E-06	8.82E-06	8.08E-06
Grnd Surface	3.23E-03	1.38E-02	3.54E-03	2.84E-03	2.65E-03	3.23E-03	2.89E-03
Ingestion	1.24E-01	5.28E-04	4.16E+00	9.96E-04	5.16E-04	7.85E-04	8.17E-04
TOTALS	1.33E-01	2.51E-02	4.19E+00	8.58E-03	7.34E-03	8.75E-03	8.17E-03

Table C-34. (continued).

Report Date: 24-Jun-91
Infant receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Ar-41	1.83E+00	7.24E+02	7.86E-12	2.44E-09	0.00E+00
Br-84	5.30E-01	1.15E+01	1.25E-13	3.87E-11	1.80E-03
Kr-85m	4.48E+00	3.38E+01	3.67E-13	1.14E-10	0.00E+00
Kr-87	1.27E+00	1.42E+02	1.54E-12	4.79E-10	0.00E+00
Kr-88	2.84E+00	1.77E+02	1.92E-12	5.96E-10	0.00E+00
Rb-89	2.53E-01	3.41E+01	3.70E-13	1.15E-10	1.80E-03
Sr-89	1.21E+03	4.64E-02	5.04E-16	1.56E-13	1.80E-03
Sr-90	2.50E+05	2.73E-04	2.96E-18	9.20E-16	1.80E-03
Sr-91	9.50E+00	2.86E-01	3.11E-15	9.64E-13	1.80E-03
Sr-92	2.71E+00	3.30E-01	3.58E-15	1.11E-12	1.80E-03
Y-91	1.40E+03	2.41E-02	2.62E-16	8.12E-14	1.80E-03
Y-92	3.54E+00	1.82E-01	1.97E-15	6.13E-13	1.80E-03
Y-93	1.02E+01	1.73E-01	1.87E-15	5.82E-13	1.80E-03
Zr-95	1.54E+03	2.63E-02	2.85E-16	8.85E-14	1.80E-03
Zr-97	1.69E+01	1.21E-01	1.32E-15	4.08E-13	1.80E-03
Nb-96	2.34E+01	1.02E-05	1.11E-19	3.45E-17	1.80E-03
Mo-99	6.60E+01	6.40E-02	6.94E-16	2.16E-13	1.80E-03
Ru-103	9.44E+02	1.58E-02	1.72E-16	5.33E-14	1.80E-03
Ru-105	4.44E+00	4.32E-02	4.69E-16	1.46E-13	1.80E-03
Ru-106	8.84E+03	4.53E-04	4.92E-18	1.53E-15	1.80E-03
Sb-129	4.32E+00	2.85E-02	3.09E-16	9.60E-14	1.80E-03
Te-131	4.20E-01	1.58E-01	1.71E-15	5.32E-13	1.80E-03
Te-131m	3.00E+01	5.55E-03	6.02E-17	1.87E-14	1.80E-03
Te-132	7.80E+01	4.32E-02	4.69E-16	1.46E-13	1.80E-03
Te-133m	9.20E-01	1.59E-01	1.73E-15	5.36E-13	1.80E-03
Te-134	7.00E-01	3.26E-01	3.54E-15	1.10E-12	1.80E-03
I-131 (ELE)	1.93E+02	4.55E+00	4.94E-14	1.53E-11	1.80E-02
I-132	2.30E+00	5.56E+00	6.04E-14	1.88E-11	1.80E-02
I-133	2.08E+01	2.27E+01	2.47E-13	7.66E-11	1.80E-02
I-134	8.77E-01	6.24E+01	6.78E-13	2.10E-10	1.80E-02
I-135	6.61E+00	4.48E+01	4.87E-13	1.51E-10	1.80E-02
Xe-129m	2.13E+02	4.02E-08	4.36E-22	1.35E-19	0.00E+00
Xe-135	9.08E+00	7.89E+01	8.57E-13	2.66E-10	0.00E+00
Xe-135m	2.56E-01	1.54E+01	1.67E-13	5.20E-11	0.00E+00
Xe-138	2.36E-01	1.60E+02	1.74E-12	5.41E-10	0.00E+00
Cs-137	2.65E+05	4.83E-04	5.25E-18	1.63E-15	1.80E-03
Cs-138	5.37E-01	1.32E+02	1.44E-12	4.46E-10	1.80E-03
Ba-139	1.38E+00	3.80E+00	4.13E-14	1.28E-11	1.80E-03
Ba-140	3.07E+02	4.64E-02	5.04E-16	1.56E-13	1.80E-03
Ba-141	3.00E-01	2.61E-01	2.83E-15	8.78E-13	1.80E-03
Ba-142	1.80E-01	1.22E-01	1.32E-15	4.11E-13	1.80E-03
La-141	3.93E+00	2.74E-01	2.97E-15	9.23E-13	1.80E-03
La-142	1.59E+00	3.63E-01	3.94E-15	1.22E-12	1.80E-03
Ce-141	7.80E+02	3.01E-02	3.27E-16	1.02E-13	1.80E-03
Ce-143	3.30E+01	8.65E-02	9.39E-16	2.92E-13	1.80E-03
Ce-144	6.82E+03	7.41E-03	8.05E-17	2.50E-14	1.80E-03
Pr-143	3.25E+02	3.11E-02	3.38E-16	1.05E-13	1.80E-03
Pr-144	2.90E-01	7.40E-03	8.04E-17	2.49E-14	1.80E-03
U-234	2.14E+09	1.97E-05	2.14E-19	6.64E-17	1.80E-03
U-235	6.17E+12	6.26E-07	6.80E-21	2.11E-18	1.80E-03
U-238	3.91E+13	5.82E-09	6.32E-23	1.96E-20	1.80E-03
TOTAL		1.66E+03			

Table C-34. (continued).

Report Date: 24-Jun-91
Infant Receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	1.84E-03	3.39E-03	2.04E-03	1.64E-03	1.55E-03	1.75E-03	1.65E-03
Br-84	4.80E-05	1.27E-04	5.07E-05	4.58E-05	4.02E-05	4.46E-05	4.32E-05
Kr-85m	1.06E-05	3.40E-05	1.17E-05	8.73E-06	7.99E-06	1.28E-05	8.38E-06
Kr-87	2.44E-04	9.01E-04	2.61E-04	2.16E-04	2.05E-04	2.34E-04	2.21E-04
Kr-88	1.04E-03	2.84E-03	1.12E-03	9.30E-04	8.85E-04	9.84E-04	9.47E-04
Rb-89	1.54E-04	3.31E-04	1.67E-04	1.40E-04	1.29E-04	1.45E-04	1.39E-04
Sr-89	2.39E-06	8.34E-05	2.15E-09	7.67E-06	1.61E-09	1.87E-09	1.74E-09
Sr-90	5.60E-06	1.29E-04	4.74E-07	4.81E-07	4.74E-07	4.31E-05	3.06E-05
Sr-91	1.49E-06	6.87E-06	1.41E-06	2.31E-06	1.05E-06	1.26E-06	1.14E-06
Sr-92	1.28E-06	1.83E-06	1.26E-06	1.74E-06	9.61E-07	1.09E-06	1.03E-06
Y-91	7.13E-07	5.23E-05	3.15E-08	4.78E-06	2.37E-08	2.70E-08	2.54E-08
Y-92	1.92E-07	3.53E-06	1.45E-07	5.43E-07	1.09E-07	1.27E-07	1.18E-07
Y-93	2.64E-07	6.34E-06	7.60E-08	9.18E-07	5.79E-08	6.88E-08	6.25E-08
Zr-95	2.82E-05	3.45E-05	2.81E-05	3.86E-05	2.13E-05	2.74E-05	2.33E-05
Zr-97	1.47E-06	8.19E-06	1.34E-06	1.16E-06	9.96E-07	1.19E-06	1.08E-06
Nb-96	2.39E-10	3.54E-10	2.19E-10	2.07E-10	1.63E-10	1.94E-10	1.77E-10
Mo-99	4.08E-07	3.87E-06	2.10E-07	7.11E-07	1.54E-07	1.92E-07	1.66E-07
Ru-103	2.73E-06	2.84E-06	2.14E-06	6.04E-06	1.67E-06	2.03E-06	1.78E-06
Ru-105	1.16E-07	3.56E-07	1.17E-07	1.42E-07	8.70E-08	1.07E-07	9.49E-08
Ru-106	1.23E-06	1.58E-05	2.83E-07	7.21E-06	2.28E-07	2.73E-07	2.68E-07
Sb-129	1.39E-07	5.12E-07	1.44E-07	1.66E-07	1.08E-07	1.28E-07	1.17E-07
Te-131	1.47E-07	6.22E-07	3.29E-07	1.44E-07	1.11E-07	1.46E-07	1.18E-07
Te-131m	9.78E-08	2.02E-07	4.57E-07	9.42E-08	6.57E-08	7.92E-08	7.08E-08
Te-132	2.63E-06	7.25E-06	8.70E-06	2.26E-06	1.99E-06	2.41E-06	2.15E-06
Te-133m	8.83E-07	1.84E-06	1.75E-06	8.15E-07	7.02E-07	8.31E-07	7.56E-07
Te-134	6.26E-07	8.77E-07	1.02E-06	5.64E-07	4.90E-07	6.28E-07	5.33E-07
I-131 (ELE)	1.25E-01	2.88E-03	4.16E+00	1.83E-03	1.23E-03	1.69E-03	1.58E-03
I-132	1.10E-04	2.86E-04	3.07E-04	1.18E-04	8.33E-05	9.82E-05	9.00E-05
I-133	1.41E-03	5.27E-03	2.31E-02	6.64E-04	6.21E-04	7.64E-04	6.79E-04
I-134	6.90E-04	1.86E-03	7.97E-04	6.24E-04	5.68E-04	6.67E-04	6.13E-04
I-135	1.25E-03	3.92E-03	2.17E-03	1.13E-03	1.02E-03	1.17E-03	1.10E-03
Xe-129m	1.80E-15	1.48E-14	1.85E-15	1.12E-15	9.66E-16	1.77E-15	7.99E-16
Xe-135	3.79E-05	1.10E-04	4.12E-05	3.18E-05	2.95E-05	4.24E-05	3.24E-05
Xe-135m	1.27E-05	2.12E-05	1.38E-05	1.11E-05	1.03E-05	1.29E-05	1.13E-05
Xe-138	3.86E-04	7.95E-04	4.16E-04	3.43E-04	3.25E-04	3.73E-04	3.48E-04
Cs-137	1.41E-05	4.15E-05	1.53E-05	1.24E-05	1.17E-05	1.41E-05	1.33E-05
Cs-138	6.91E-04	1.64E-03	7.49E-04	6.51E-04	5.77E-04	6.52E-04	6.20E-04
Ba-139	6.39E-07	2.31E-05	3.47E-07	2.15E-06	2.39E-07	3.72E-07	2.49E-07
Ba-140	1.00E-05	4.06E-05	1.07E-05	8.79E-06	8.17E-06	9.66E-06	8.85E-06
Ba-141	4.87E-07	1.46E-06	5.23E-07	4.77E-07	3.89E-07	4.88E-07	4.22E-07
Ba-142	2.26E-07	4.73E-07	2.48E-07	2.09E-07	1.84E-07	2.21E-07	1.98E-07
La-141	1.13E-07	3.54E-06	3.71E-08	3.81E-07	1.42E-07	3.19E-08	3.02E-08
La-142	2.59E-06	5.51E-06	2.71E-06	2.48E-06	2.16E-06	2.38E-06	2.30E-06
Ce-141	6.75E-07	1.64E-06	5.84E-07	1.43E-06	3.80E-07	6.56E-07	3.76E-07
Ce-143	3.99E-07	2.97E-06	2.72E-07	8.55E-07	1.93E-07	2.64E-07	2.06E-07
Ce-144	1.32E-05	1.65E-04	7.79E-07	9.11E-05	1.62E-06	2.98E-06	3.25E-06
Pr-143	1.33E-07	5.80E-06	2.61E-14	7.96E-07	1.94E-14	2.32E-14	2.11E-14
Pr-144	6.91E-10	3.51E-08	5.66E-10	1.85E-09	4.43E-10	4.99E-10	4.75E-10
U-234	1.43E-06	9.57E-09	3.47E-10	1.16E-05	1.40E-10	9.96E-07	6.57E-08
U-235	5.10E-08	1.38E-08	9.69E-09	3.41E-07	6.46E-09	3.93E-08	8.59E-09
U-238	4.05E-10	6.40E-09	1.74E-11	3.12E-09	1.17E-11	2.83E-10	2.91E-11

Table C-34. (continued).

Report Date: 24-Jun-91
Infant receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	1.84E-03	3.39E-03	2.04E-03	1.64E-03	1.55E-03	1.75E-03	1.65E-03
Br-84	4.50E-05	1.03E-04	4.81E-05	4.00E-05	3.81E-05	4.23E-05	4.10E-05
Kr-85m	1.06E-05	3.40E-05	1.17E-05	8.73E-06	7.99E-06	1.28E-05	8.38E-06
Kr-87	2.44E-04	9.01E-04	2.61E-04	2.16E-04	2.05E-04	2.34E-04	2.21E-04
Kr-88+D	1.04E-03	2.84E-03	1.12E-03	9.30E-04	8.85E-04	9.84E-04	9.47E-04
Rb-89	1.49E-04	3.03E-04	1.62E-04	1.32E-04	1.25E-04	1.41E-04	1.35E-04
Sr-89	1.28E-11	7.80E-08	1.43E-11	1.13E-11	1.06E-11	1.24E-11	1.15E-11
Sr-90+D	0.00E+00	8.90E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	5.80E-07	1.34E-06	6.41E-07	5.10E-07	4.77E-07	5.69E-07	5.18E-07
Sr-92	8.77E-07	1.32E-06	9.67E-07	7.80E-07	7.37E-07	8.33E-07	7.88E-07
Y-91	1.73E-10	4.23E-08	1.93E-10	1.54E-10	1.45E-10	1.65E-10	1.56E-10
Y-92	9.22E-08	9.43E-07	1.02E-07	8.11E-08	7.62E-08	8.87E-08	8.25E-08
Y-93	3.20E-08	6.70E-07	3.48E-08	2.82E-08	2.66E-08	3.14E-08	2.86E-08
Zr-95+D	1.26E-07	1.76E-07	1.40E-07	1.11E-07	1.04E-07	1.24E-07	1.13E-07
Zr-97+D	3.82E-07	9.33E-07	4.21E-07	3.36E-07	3.14E-07	3.77E-07	3.42E-07
Nb-96	5.03E-11	7.39E-11	5.58E-11	4.44E-11	4.17E-11	4.91E-11	4.48E-11
Mo-99+D	1.96E-08	9.85E-08	2.17E-08	1.72E-08	1.60E-08	1.99E-08	1.74E-08
Ru-103+D	1.47E-08	2.06E-08	1.59E-08	1.28E-08	1.19E-08	1.50E-08	1.31E-08
Ru-105+D	6.73E-08	1.37E-07	7.42E-08	5.89E-08	5.50E-08	6.76E-08	6.00E-08
Ru-106+D	1.83E-10	2.23E-09	2.00E-10	1.61E-10	1.50E-10	1.83E-10	1.64E-10
Sb-129+D	8.54E-08	1.88E-07	9.44E-08	7.55E-08	7.08E-08	8.32E-08	7.65E-08
Te-131	1.31E-07	4.94E-07	1.44E-07	1.13E-07	1.05E-07	1.38E-07	1.12E-07
Te-131m+D	1.69E-08	2.71E-08	1.87E-08	1.48E-08	1.39E-08	1.67E-08	1.50E-08
Te-132+D	2.21E-07	3.60E-07	2.44E-07	1.94E-07	1.81E-07	2.19E-07	1.96E-07
Te-133m+D	7.64E-07	1.41E-06	8.42E-07	6.72E-07	6.31E-07	7.46E-07	6.79E-07
Te-134	5.56E-07	8.04E-07	6.11E-07	4.81E-07	4.49E-07	5.74E-07	4.87E-07
I-131 (ELE)	3.34E-06	6.16E-06	3.62E-06	2.87E-06	2.66E-06	3.53E-06	2.96E-06
I-132	2.54E-05	4.17E-05	2.80E-05	2.24E-05	2.10E-05	2.48E-05	2.27E-05
I-133	2.68E-05	6.08E-05	2.92E-05	2.35E-05	2.19E-05	2.69E-05	2.39E-05
I-134	3.31E-04	5.54E-04	3.67E-04	2.93E-04	2.73E-04	3.21E-04	2.95E-04
I-135	1.44E-04	2.36E-04	1.58E-04	1.28E-04	1.21E-04	1.37E-04	1.29E-04
Xe-129m	1.80E-15	1.48E-14	1.85E-15	1.12E-15	9.66E-16	1.77E-15	7.99E-16
Xe-135	3.79E-05	1.10E-04	4.12E-05	3.18E-05	2.95E-05	4.24E-05	3.24E-05
Xe-135m	1.27E-05	2.12E-05	1.38E-05	1.11E-05	1.03E-05	1.29E-05	1.13E-05
Xe-138	3.86E-04	7.95E-04	4.16E-04	3.43E-04	3.25E-04	3.73E-04	3.48E-04
Cs-137+D	5.38E-10	9.71E-10	5.92E-10	4.71E-10	4.41E-10	5.34E-10	4.81E-10
Cs-138	6.46E-04	1.35E-03	7.07E-04	5.75E-04	5.45E-04	6.16E-04	5.85E-04
Ba-139	2.63E-07	1.05E-05	2.91E-07	2.18E-07	2.00E-07	3.11E-07	2.09E-07
Ba-140+D	2.69E-07	4.68E-07	2.93E-07	2.38E-07	2.25E-07	2.59E-07	2.42E-07
Ba-141	4.59E-07	1.26E-06	5.03E-07	4.00E-07	3.74E-07	4.70E-07	4.06E-07
Ba-142	2.19E-07	4.44E-07	2.43E-07	1.92E-07	1.80E-07	2.16E-07	1.94E-07
La-141	2.35E-08	8.10E-07	2.58E-08	2.08E-08	1.98E-08	2.22E-08	2.10E-08
La-142	2.19E-06	3.77E-06	2.33E-06	1.95E-06	1.86E-06	2.05E-06	1.98E-06
Ce-141	4.47E-09	1.38E-08	4.99E-09	3.60E-09	3.26E-09	5.62E-09	3.26E-09
Ce-143	4.39E-08	1.55E-07	4.79E-08	3.69E-08	3.42E-08	4.62E-08	3.69E-08
Ce-144+D	7.61E-10	2.87E-08	8.32E-10	6.50E-10	6.08E-10	8.08E-10	6.28E-10
Pr-143	5.52E-16	2.47E-08	6.10E-16	4.85E-16	4.54E-16	5.41E-16	4.93E-16
Pr-144	5.09E-10	2.81E-08	5.49E-10	4.52E-10	4.29E-10	4.83E-10	4.61E-10
U-234	5.79E-15	2.47E-14	4.60E-15	3.11E-15	2.62E-15	5.37E-15	2.09E-15
U-235+D	2.00E-13	3.04E-13	2.20E-13	1.62E-13	1.48E-13	2.43E-13	1.53E-13
U-238+D	2.21E-16	1.47E-14	2.50E-16	1.85E-16	1.71E-16	2.50E-16	1.66E-16
TOTALS	4.95E-03	1.08E-02	5.41E-03	4.40E-03	4.17E-03	4.73E-03	4.46E-03

Table C-34. (continued).

Report Date: 24-Jun-91
Infant receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	5.33E-07	0.00E+00	0.00E+00	3.61E-06	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	6.73E-07	0.00E+00	0.00E+00	4.55E-06	0.00E+00	0.00E+00	0.00E+00
Sr-89	9.15E-07	0.00E+00	0.00E+00	7.67E-06	0.00E+00	0.00E+00	0.00E+00
Sr-90	6.47E-08	5.39E-09	5.39E-09	1.24E-08	5.39E-09	5.13E-07	3.67E-07
Sr-91	2.14E-07	0.00E+00	0.00E+00	1.19E-06	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.35E-07	0.00E+00	0.00E+00	7.21E-07	0.00E+00	0.00E+00	0.00E+00
Y-91	5.66E-07	0.00E+00	0.00E+00	4.76E-06	0.00E+00	0.00E+00	0.00E+00
Y-92	6.02E-08	0.00E+00	0.00E+00	4.27E-07	0.00E+00	0.00E+00	0.00E+00
Y-93	1.93E-07	0.00E+00	0.00E+00	8.56E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	2.59E-06	1.98E-07	3.90E-07	1.65E-05	6.74E-07	1.30E-06	5.09E-07
Zr-97	2.59E-07	0.00E+00	0.00E+00	9.70E-08	0.00E+00	0.00E+00	0.00E+00
Nb-96	1.09E-11	0.00E+00	0.00E+00	3.22E-11	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.23E-07	0.00E+00	0.00E+00	5.46E-07	0.00E+00	0.00E+00	0.00E+00
Ru-103	6.86E-07	4.06E-08	6.86E-08	4.37E-06	1.10E-07	7.49E-08	6.24E-08
Ru-105	9.45E-09	0.00E+00	0.00E+00	4.84E-08	0.00E+00	0.00E+00	0.00E+00
Ru-106	8.78E-07	1.34E-08	1.52E-08	6.99E-06	1.79E-08	1.79E-08	2.13E-08
Sb-129	8.66E-09	0.00E+00	0.00E+00	5.02E-08	0.00E+00	0.00E+00	0.00E+00
Te-131	8.34E-09	0.00E+00	1.77E-07	2.53E-08	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.63E-08	0.00E+00	3.55E-07	2.40E-08	0.00E+00	0.00E+00	0.00E+00
Te-132	1.77E-07	0.00E+00	5.30E-06	1.45E-07	0.00E+00	0.00E+00	0.00E+00
Te-133m	3.22E-08	0.00E+00	8.14E-07	6.62E-08	0.00E+00	0.00E+00	0.00E+00
Te-134	1.74E-08	0.00E+00	3.48E-07	3.83E-08	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	6.19E-04	2.70E-06	2.07E-02	6.29E-05	2.96E-06	3.96E-06	4.06E-06
I-132	1.10E-05	8.67E-07	1.98E-04	3.09E-05	1.54E-06	1.32E-06	1.32E-06
I-133	6.55E-05	0.00E+00	2.18E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	3.66E-06	0.00E+00	3.66E-05	1.73E-05	0.00E+00	0.00E+00	0.00E+00
I-135	2.63E-05	0.00E+00	7.41E-04	3.83E-05	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	1.24E-08	9.44E-09	1.15E-08	1.91E-08	1.15E-08	1.24E-08	1.62E-08
Cs-138	6.22E-06	0.00E+00	0.00E+00	4.17E-05	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.25E-07	0.00E+00	0.00E+00	1.89E-06	0.00E+00	0.00E+00	0.00E+00
Ba-140	8.92E-08	0.00E+00	0.00E+00	1.56E-07	0.00E+00	2.20E-07	1.19E-07
Ba-141	1.03E-08	0.00E+00	0.00E+00	6.11E-08	0.00E+00	0.00E+00	0.00E+00
Ba-142	2.34E-09	0.00E+00	0.00E+00	1.30E-08	0.00E+00	0.00E+00	0.00E+00
La-141	7.89E-08	0.00E+00	0.00E+00	3.51E-07	1.14E-07	0.00E+00	0.00E+00
La-142	4.26E-08	0.00E+00	0.00E+00	2.13E-07	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.37E-07	0.00E+00	0.00E+00	1.01E-06	0.00E+00	0.00E+00	0.00E+00
Ce-143	1.48E-07	0.00E+00	0.00E+00	6.46E-07	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.15E-05	3.80E-08	4.55E-08	9.05E-05	8.07E-07	1.40E-06	1.61E-06
Pr-143	1.21E-07	0.00E+00	0.00E+00	7.96E-07	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.66E-10	0.00E+00	0.00E+00	1.38E-09	0.00E+00	0.00E+00	0.00E+00
U-234	1.37E-06	0.00E+00	0.00E+00	1.16E-05	0.00E+00	0.00E+00	0.00E+00
U-235	4.01E-08	0.00E+00	0.00E+00	3.34E-07	0.00E+00	0.00E+00	0.00E+00
U-238	3.73E-10	0.00E+00	0.00E+00	3.10E-09	0.00E+00	0.00E+00	0.00E+00
TOTALS	7.54E-04	3.87E-06	2.38E-02	3.51E-04	6.24E-06	8.82E-06	8.08E-06

Table C-34. (continued).

Report Date: 24-Jun-91
Infant receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	2.43E-06	2.34E-05	2.62E-06	2.16E-06	2.05E-06	2.30E-06	2.20E-06
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.13E-06	2.87E-05	4.52E-06	3.67E-06	3.47E-06	3.93E-06	3.73E-06
Sr-89	1.92E-09	8.33E-05	2.14E-09	1.70E-09	1.60E-09	1.85E-09	1.73E-09
Sr-90+D	0.00E+00	1.29E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	6.98E-07	5.53E-06	7.70E-07	6.14E-07	5.75E-07	6.87E-07	6.24E-07
Sr-92	2.67E-07	5.12E-07	2.96E-07	2.38E-07	2.25E-07	2.54E-07	2.40E-07
Y-91	2.82E-08	5.22E-05	3.13E-08	2.50E-08	2.36E-08	2.68E-08	2.53E-08
Y-92	3.92E-08	2.59E-06	4.34E-08	3.46E-08	3.25E-08	3.79E-08	3.51E-08
Y-93	3.78E-08	5.67E-06	4.12E-08	3.33E-08	3.13E-08	3.74E-08	3.38E-08
Zr-95+D	2.48E-05	3.41E-05	2.75E-05	2.19E-05	2.05E-05	2.43E-05	2.22E-05
Zr-97+D	8.29E-07	7.26E-06	9.14E-07	7.29E-07	6.82E-07	8.18E-07	7.42E-07
Nb-96	1.47E-10	2.80E-10	1.63E-10	1.30E-10	1.22E-10	1.44E-10	1.32E-10
Mo-99+D	1.70E-07	3.77E-06	1.88E-07	1.48E-07	1.38E-07	1.72E-07	1.49E-07
Ru-103+D	1.88E-06	2.76E-06	2.03E-06	1.63E-06	1.52E-06	1.92E-06	1.67E-06
Ru-105+D	3.93E-08	2.18E-07	4.31E-08	3.43E-08	3.20E-08	3.95E-08	3.49E-08
Ru-106+D	2.11E-07	1.57E-05	2.31E-07	1.85E-07	1.73E-07	2.12E-07	1.89E-07
Sb-129+D	4.53E-08	3.23E-07	5.01E-08	4.01E-08	3.75E-08	4.44E-08	4.10E-08
Te-131	7.27E-09	1.28E-07	8.02E-09	6.24E-09	5.79E-09	7.74E-09	6.20E-09
Te-131m+D	6.31E-08	1.75E-07	6.98E-08	5.54E-08	5.18E-08	6.26E-08	5.58E-08
Te-132+D	2.21E-06	6.89E-06	2.44E-06	1.92E-06	1.81E-06	2.19E-06	1.95E-06
Te-133m+D	8.62E-08	4.25E-07	9.56E-08	7.60E-08	7.12E-08	8.48E-08	7.68E-08
Te-134	5.21E-08	7.23E-08	5.73E-08	4.50E-08	4.18E-08	5.40E-08	4.54E-08
I-131 (ELE)	8.93E-04	2.34E-03	9.68E-04	7.67E-04	7.12E-04	9.46E-04	7.89E-04
I-132	7.37E-05	2.44E-04	8.14E-05	6.47E-05	6.08E-05	7.21E-05	6.60E-05
I-133	7.32E-04	5.20E-03	7.99E-04	6.41E-04	5.99E-04	7.37E-04	6.55E-04
I-134	3.55E-04	1.31E-03	3.94E-04	3.14E-04	2.95E-04	3.46E-04	3.18E-04
I-135	1.08E-03	3.69E-03	1.18E-03	9.61E-04	9.02E-04	1.03E-03	9.69E-04
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	1.27E-05	4.03E-05	1.40E-05	1.11E-05	1.04E-05	1.26E-05	1.13E-05
Cs-138	3.84E-05	2.86E-04	4.18E-05	3.41E-05	3.24E-05	3.66E-05	3.45E-05
Ba-139	5.11E-08	1.26E-05	5.65E-08	4.20E-08	3.85E-08	6.06E-08	4.01E-08
Ba-140+D	9.49E-06	4.01E-05	1.04E-05	8.40E-06	7.94E-06	9.18E-06	8.49E-06
Ba-141	1.76E-08	1.97E-07	1.93E-08	1.53E-08	1.42E-08	1.81E-08	1.55E-08
Ba-142	4.87E-09	2.88E-08	5.40E-09	4.27E-09	4.00E-09	4.82E-09	4.29E-09
La-141	1.03E-08	2.73E-06	1.13E-08	9.12E-09	8.63E-09	9.72E-09	9.20E-09
La-142	3.54E-07	1.74E-06	3.79E-07	3.15E-07	3.01E-07	3.34E-07	3.22E-07
Ce-141	5.19E-07	1.63E-06	5.79E-07	4.15E-07	3.77E-07	6.50E-07	3.73E-07
Ce-143	2.05E-07	2.81E-06	2.24E-07	1.72E-07	1.59E-07	2.18E-07	1.69E-07
Ce-144+D	6.48E-07	1.65E-04	7.13E-07	5.46E-07	5.07E-07	7.02E-07	5.14E-07
Pr-143	2.31E-14	5.78E-06	2.54E-14	2.03E-14	1.90E-14	2.26E-14	2.06E-14
Pr-144	1.61E-11	7.07E-09	1.74E-11	1.43E-11	1.35E-11	1.53E-11	1.45E-11
U-234	1.21E-09	9.57E-09	3.47E-10	2.62E-10	1.40E-10	4.44E-10	1.38E-10
U-235+D	9.00E-09	1.38E-08	9.69E-09	7.11E-09	6.46E-09	1.07E-08	6.66E-09
U-238+D	1.61E-11	6.40E-09	1.74E-11	1.28E-11	1.17E-11	1.78E-11	1.12E-11
TOTALS	3.23E-03	1.38E-02	3.54E-03	2.84E-03	2.65E-03	3.23E-03	2.89E-03

Table C-34. (continued).

Report Date: 24-Jun-91
Infant receptor

9/7-10/14/60 Episodic Release
IET #23 (FEET #2)

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	7.93E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.15E-38	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.48E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	5.53E-06	4.68E-07	4.68E-07	4.68E-07	4.68E-07	4.26E-05	3.03E-05
Sr-91	2.86E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.39E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	1.19E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	3.98E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	5.64E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	7.01E-07	4.62E-08	2.67E-08	4.62E-08	7.71E-08	1.61E-06	4.42E-07
Zr-97	2.52E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	3.08E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	9.57E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.52E-07	1.94E-08	2.17E-08	2.17E-08	2.96E-08	2.76E-08	3.16E-08
Ru-105	1.61E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.41E-07	3.50E-08	3.66E-08	3.66E-08	3.66E-08	4.29E-08	5.72E-08
Sb-129	1.80E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	2.18E-29	0.00E+00	3.60E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.45E-09	0.00E+00	1.36E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	2.44E-08	0.00E+00	7.24E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	6.02E-19	0.00E+00	1.27E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	7.34E-22	0.00E+00	1.15E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	1.23E-01	5.26E-04	4.14E+00	9.94E-04	5.14E-04	7.37E-04	7.83E-04
I-132	2.49E-10	2.08E-11	5.69E-09	4.15E-11	3.32E-11	2.91E-11	3.32E-11
I-133	5.90E-04	0.00E+00	2.01E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	2.15E-16	0.00E+00	2.61E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.63E-06	0.00E+00	8.81E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	1.42E-06	1.13E-06	1.35E-06	1.28E-06	1.35E-06	1.49E-06	1.91E-06
Cs-138	1.42E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	4.48E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.52E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	1.08E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	2.45E-53	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	7.93E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	3.73E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.47E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	2.89E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.02E-06	2.04E-08	2.04E-08	2.04E-08	3.06E-07	8.78E-07	1.12E-06
Pr-143	1.18E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	7.00E-39	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-234	6.32E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.96E-07	6.56E-08
U-235	1.93E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.86E-08	1.93E-09
U-238	1.65E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.66E-10	1.79E-11
TOTALS	1.24E-01	5.28E-04	4.16E+00	9.96E-04	5.16E-04	7.85E-04	8.17E-04

Table C-35. Dose evaluation for FPFRT-8 (adult)

Report Date: 19-May-91
File Name: ACC8A.WK1

Episodic Release

Episode Description..... FPFRT-8
Date..... 9/18/58
Receptor Location..... Mud Lake
Receptor Designation..... Adult
Release Location..... GRID III

Input Parameters

5.83E-10 : Episodic Mesoscale Dispersion Factor (hr/m**3)
2.29E+01 : Exposure Period (hr)
9.59E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 02-Jun-91

9/18/58 Episodic Release
FPFRT-8

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	3.61E-02	6.95E-02	3.99E-02	3.18E-02	2.98E-02	3.53E-02	3.23E-02
Inhalation	2.25E+00	0.00E+00	7.04E-01	1.53E+01	0.00E+00	7.20E+00	9.54E-01
Grnd Surface	8.52E+00	6.60E+01	9.42E+00	7.49E+00	7.00E+00	8.39E+00	7.61E+00
Ingestion	2.82E+00	0.00E+00	3.23E+01	4.24E-01	0.00E+00	9.51E+00	4.40E+00
TOTALS	1.36E+01	6.61E+01	4.25E+01	2.33E+01	7.03E+00	2.51E+01	1.30E+01

Table C-35. (continued).

Report Date: 02-Jun-91
Adult receptor

9/18/58 Episodic Release
FPFRT-8

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Sr-89	1.21E+03	1.83E+01	4.66E-10	1.07E-08	1.80E-03
Sr-90	2.50E+05	2.72E-01	6.92E-12	1.59E-10	1.80E-03
Y-91	1.40E+03	2.21E+01	5.63E-10	1.29E-08	1.80E-03
Zr-95	1.54E+03	3.46E+01	8.81E-10	2.02E-08	1.80E-03
Ru-103	9.44E+02	3.28E+00	8.35E-11	1.91E-09	1.80E-03
Ru-106	8.84E+03	1.34E-01	3.41E-12	7.81E-11	1.80E-03
I-131 (ELE)	1.93E+02	1.13E+00	2.88E-11	6.59E-10	1.80E-02
I-132	2.30E+00	8.38E-04	2.13E-14	4.89E-13	1.80E-02
Cs-137	2.65E+05	5.39E-01	1.37E-11	3.14E-10	1.80E-03
Ba-140	3.07E+02	6.16E+00	1.57E-10	3.59E-09	1.80E-03
Ce-141	7.80E+02	8.49E+00	2.16E-10	4.95E-09	1.80E-03
Ce-144	6.82E+03	3.17E+00	8.07E-11	1.85E-09	1.80E-03
Pr-143	3.25E+02	2.08E+00	5.30E-11	1.21E-09	1.80E-03
Pr-144	2.90E-01	3.17E+00	8.07E-11	1.85E-09	1.80E-03
TOTAL		1.03E+02			

Table C-35. (continued).

Report Date: 02-Jun-91
Adult receptor

9/18/58 Episodic Release
FPFRT-8

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	6.97E-01	5.69E+00	1.47E-04	3.17E+00	1.10E-04	1.27E-04	1.19E-04
Sr-90	9.36E-01	2.22E+01	0.00E+00	1.67E+00	0.00E+00	9.09E+00	3.98E+00
Y-91	5.55E-01	8.30E+00	5.00E-03	4.58E+00	3.76E-03	4.28E-03	4.03E-03
Zr-95	6.25E+00	7.81E+00	6.31E+00	5.01E+00	4.69E+00	1.27E+01	6.02E+00
Ru-103	9.32E-02	1.00E-01	7.36E-02	1.67E-01	5.50E-02	6.93E-02	6.05E-02
Ru-106	4.98E-02	8.04E-01	1.18E-02	2.87E-01	8.87E-03	1.09E-02	9.69E-03
I-131 (ELE)	9.98E-01	1.01E-01	3.26E+01	3.30E-02	3.07E-02	4.08E-02	3.40E-02
I-132	1.01E-06	1.73E-06	3.90E-06	1.22E-06	7.06E-07	8.36E-07	7.64E-07
Cs-137	2.90E+00	7.78E+00	3.12E+00	2.58E+00	2.00E+00	2.86E+00	2.62E+00
Ba-140	2.71E-01	9.32E-01	2.45E-01	2.20E-01	1.87E-01	2.47E-01	2.17E-01
Ce-141	8.76E-02	7.99E-02	2.85E-02	3.19E-01	1.85E-02	3.20E-02	1.83E-02
Ce-144	7.65E-01	1.22E+01	5.28E-02	5.18E+00	3.75E-02	5.20E-02	3.81E-02
Pr-143	1.44E-02	6.71E-02	3.02E-10	5.58E-02	2.25E-10	2.68E-10	2.45E-10
Pr-144	1.12E-04	2.13E-03	4.08E-05	6.54E-04	3.19E-05	3.59E-05	3.42E-05

Table C-35. (continued).

Report Date: 19-May-91
Adult receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	8.72E-07	5.33E-03	9.74E-07	7.71E-07	7.24E-07	8.44E-07	7.82E-07
Sr-90+D	0.00E+00	1.53E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.75E-05	6.71E-03	3.06E-05	2.44E-05	2.31E-05	2.61E-05	2.47E-05
Zr-95+D	2.88E-02	4.01E-02	3.19E-02	2.53E-02	2.37E-02	2.82E-02	2.58E-02
Ru-103+D	5.28E-04	7.40E-04	5.71E-04	4.60E-04	4.27E-04	5.39E-04	4.71E-04
Ru-106+D	9.35E-06	1.14E-04	1.02E-05	8.21E-06	7.67E-06	9.35E-06	8.38E-06
I-131 (ELE)	1.43E-04	2.64E-04	1.55E-04	1.23E-04	1.14E-04	1.52E-04	1.27E-04
I-132	6.63E-07	1.09E-06	7.30E-07	5.85E-07	5.46E-07	6.46E-07	5.90E-07
Cs-137+D	1.04E-04	1.87E-04	1.14E-04	9.08E-05	8.51E-05	1.03E-04	9.29E-05
Ba-140+D	6.18E-03	1.07E-02	6.73E-03	5.46E-03	5.16E-03	5.95E-03	5.56E-03
Ce-141	2.18E-04	6.71E-04	2.43E-04	1.75E-04	1.59E-04	2.74E-04	1.59E-04
Ce-144+D	5.63E-05	2.12E-03	6.16E-05	4.81E-05	4.50E-05	5.98E-05	4.65E-05
Pr-143	6.39E-12	2.86E-04	7.06E-12	5.61E-12	5.25E-12	6.26E-12	5.71E-12
Pr-144	3.77E-05	2.08E-03	4.07E-05	3.35E-05	3.18E-05	3.58E-05	3.41E-05
TOTALS	3.61E-02	6.95E-02	3.99E-02	3.18E-02	2.98E-02	3.53E-02	3.23E-02

Table C-35. (continued).

Report Date: 19-May-91
Adult receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	3.79E-01	0.00E+00	0.00E+00	3.17E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.98E-01	0.00E+00	0.00E+00	1.67E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	5.44E-01	0.00E+00	0.00E+00	4.57E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	3.68E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.16E+00	9.29E-01
Ru-103	1.43E-02	0.00E+00	0.00E+00	1.08E-01	0.00E+00	0.00E+00	0.00E+00
Ru-106	3.30E-02	0.00E+00	0.00E+00	2.77E-01	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	2.02E-02	0.00E+00	6.95E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	1.55E-07	0.00E+00	2.95E-06	4.69E-07	0.00E+00	0.00E+00	0.00E+00
Cs-137	9.64E-03	0.00E+00	8.74E-03	9.94E-03	0.00E+00	8.74E-03	9.34E-03
Ba-140	1.24E-02	0.00E+00	0.00E+00	2.17E-02	0.00E+00	3.07E-02	1.65E-02
Ce-141	4.03E-02	0.00E+00	0.00E+00	2.99E-01	0.00E+00	0.00E+00	0.00E+00
Ce-144	6.20E-01	0.00E+00	0.00E+00	5.14E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	8.49E-03	0.00E+00	0.00E+00	5.58E-02	0.00E+00	0.00E+00	0.00E+00
Pr-144	7.44E-05	0.00E+00	0.00E+00	6.20E-04	0.00E+00	0.00E+00	0.00E+00
TOTALS	2.25E+00	0.00E+00	7.04E-01	1.53E+01	0.00E+00	7.20E+00	9.54E-01

Table C-35. (continued).

Report Date: 26-Jun-91
Adult receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	1.31E-04	5.69E+00	1.46E-04	1.16E-04	1.09E-04	1.27E-04	1.18E-04
Sr-90+D	0.00E+00	2.22E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	4.47E-03	8.29E+00	4.97E-03	3.97E-03	3.74E-03	4.25E-03	4.01E-03
Zr-95+D	5.66E+00	7.77E+00	6.27E+00	4.98E+00	4.66E+00	5.55E+00	5.07E+00
Ru-103+D	6.75E-02	9.92E-02	7.30E-02	5.87E-02	5.46E-02	6.88E-02	6.00E-02
Ru-106+D	1.08E-02	8.04E-01	1.18E-02	9.48E-03	8.86E-03	1.09E-02	9.68E-03
I-131 (ELE)	3.83E-02	1.01E-01	4.16E-02	3.29E-02	3.05E-02	4.06E-02	3.39E-02
I-132	1.92E-06	6.36E-06	2.12E-06	1.69E-06	1.58E-06	1.88E-06	1.72E-06
Cs-137+D	2.45E+00	7.78E+00	2.69E+00	2.15E+00	2.00E+00	2.43E+00	2.19E+00
Ba-140+D	2.18E-01	9.21E-01	2.38E-01	1.93E-01	1.82E-01	2.11E-01	1.95E-01
Ce-141	2.53E-02	7.92E-02	2.82E-02	2.02E-02	1.84E-02	3.17E-02	1.82E-02
Ce-144+D	4.80E-02	1.22E+01	5.27E-02	4.04E-02	3.75E-02	5.19E-02	3.80E-02
Pr-143	2.67E-10	6.68E-02	2.94E-10	2.35E-10	2.20E-10	2.62E-10	2.39E-10
Pr-144	1.19E-06	5.24E-04	1.29E-06	1.06E-06	1.00E-06	1.14E-06	1.08E-06
TOTALS	8.52E+00	6.60E+01	9.42E+00	7.49E+00	7.00E+00	8.39E+00	7.61E+00

Table C-35. (continued).

Report Date: 19-May-91
Adult receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	3.19E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	7.38E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.09E+00	3.98E+00
Y-91	7.06E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	2.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.08E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	6.05E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	9.39E-01	0.00E+00	3.19E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	3.15E-11	0.00E+00	7.74E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	4.42E-01	0.00E+00	4.24E-01	4.24E-01	0.00E+00	4.24E-01	4.24E-01
Ba-140	3.48E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	2.17E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	9.71E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	5.93E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	3.32E-32	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	2.82E+00	0.00E+00	3.23E+01	4.24E-01	0.00E+00	9.51E+00	4.40E+00

Table C-36. Dose evaluation for FPFRT-8 (infant)

Report Date: 18-May-91
File Name: ACC8B.WK1

Episodic Release

Episode Description..... FPFRT-8
Date..... 9/18/58
Receptor Location..... Mud Lake
Receptor Designation..... Infant
Release Location..... GRID III

Input Parameters

5.83E-10 : Episodic Mesoscale Dispersion Factor (hr/m**3)
2.29E+01 : Exposure Period (hr)
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 02-Jun-91

9/18/58 Episodic Release
FPFRT-8

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	3.61E-02	6.95E-02	3.99E-02	3.18E-02	2.98E-02	3.53E-02	3.23E-02
Inhalation	1.72E+00	5.28E-02	9.86E-01	1.23E+01	2.21E-01	4.99E-01	3.08E-01
Grnd Surface	8.52E+00	6.60E+01	9.42E+00	7.49E+00	7.00E+00	8.39E+00	7.61E+00
Ingestion	6.88E+00	3.37E-01	1.78E+02	3.84E-01	4.06E-01	8.09E+00	5.81E+00
TOTALS	1.71E+01	6.64E+01	1.89E+02	2.02E+01	7.66E+00	1.70E+01	1.38E+01

Table C-36. (continued).

Report Date: 18-May-91
Infant receptor

9/18/58 Episodic Release
FPFRT-8

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Sr-89	1.21E+03	1.83E+01	4.66E-10	1.07E-08	1.80E-03
Sr-90	2.50E+05	2.72E-01	6.92E-12	1.59E-10	1.80E-03
Y-91	1.40E+03	2.21E+01	5.63E-10	1.29E-08	1.80E-03
Zr-95	1.54E+03	3.46E+01	8.81E-10	2.02E-08	1.80E-03
Ru-103	9.44E+02	3.28E+00	8.35E-11	1.91E-09	1.80E-03
Ru-106	8.84E+03	1.34E-01	3.41E-12	7.81E-11	1.80E-03
I-131 (ELE)	1.93E+02	1.13E+00	2.88E-11	6.59E-10	1.80E-02
I-132	2.30E+00	8.38E-04	2.13E-14	4.89E-13	1.80E-02
Cs-137	2.65E+05	5.39E-01	1.37E-11	3.14E-10	1.80E-03
Ba-140	3.07E+02	6.16E+00	1.57E-10	3.59E-09	1.80E-03
Ce-141	7.80E+02	8.49E+00	2.16E-10	4.95E-09	1.80E-03
Ce-144	6.82E+03	3.17E+00	8.07E-11	1.85E-09	1.80E-03
Pr-143	3.25E+02	2.08E+00	5.30E-11	1.21E-09	1.80E-03
Pr-144	2.90E-01	3.17E+00	8.07E-11	1.85E-09	1.80E-03
TOTAL		1.03E+02			

Table C-36. (continued).

Report Date: 02-Jun-91
Infant Receptor

9/18/58 Episodic Release
FPFRT-8

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	1.63E-01	5.69E+00	1.47E-04	5.24E-01	1.10E-04	1.27E-04	1.19E-04
Sr-90	9.65E-01	2.23E+01	8.17E-02	8.29E-02	8.17E-02	7.43E+00	5.28E+00
Y-91	1.13E-01	8.30E+00	5.00E-03	7.59E-01	3.76E-03	4.28E-03	4.03E-03
Zr-95	6.44E+00	7.87E+00	6.40E+00	8.79E+00	4.86E+00	6.24E+00	5.31E+00
Ru-103	9.82E-02	1.02E-01	7.68E-02	2.17E-01	6.00E-02	7.30E-02	6.38E-02
Ru-106	6.30E-02	8.06E-01	1.45E-02	3.69E-01	1.17E-02	1.40E-02	1.37E-02
I-131 (ELE)	5.35E+00	1.24E-01	1.79E+02	7.84E-02	5.28E-02	7.26E-02	6.78E-02
I-132	1.14E-06	1.75E-06	6.09E-06	1.56E-06	7.46E-07	8.70E-07	7.98E-07
Cs-137	2.72E+00	8.00E+00	2.95E+00	2.40E+00	2.26E+00	2.72E+00	2.56E+00
Ba-140	2.29E-01	9.32E-01	2.45E-01	2.02E-01	1.87E-01	2.22E-01	2.03E-01
Ce-141	3.29E-02	7.99E-02	2.85E-02	6.98E-02	1.85E-02	3.20E-02	1.83E-02
Ce-144	9.78E-01	1.22E+01	5.77E-02	6.74E+00	1.20E-01	2.21E-01	2.40E-01
Pr-143	1.54E-03	6.71E-02	3.02E-10	9.21E-03	2.25E-10	2.68E-10	2.45E-10
Pr-144	5.01E-05	2.13E-03	4.08E-05	1.36E-04	3.19E-05	3.59E-05	3.42E-05

Table C-36. (continued).

Report Date: 18-May-91
Infant receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	8.72E-07	5.33E-03	9.74E-07	7.71E-07	7.24E-07	8.44E-07	7.82E-07
Sr-90+D	0.00E+00	1.53E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.75E-05	6.71E-03	3.06E-05	2.44E-05	2.31E-05	2.61E-05	2.47E-05
Zr-95+D	2.88E-02	4.01E-02	3.19E-02	2.53E-02	2.37E-02	2.82E-02	2.58E-02
Ru-103+D	5.28E-04	7.40E-04	5.71E-04	4.60E-04	4.27E-04	5.39E-04	4.71E-04
Ru-106+D	9.35E-06	1.14E-04	1.02E-05	8.21E-06	7.67E-06	9.35E-06	8.38E-06
I-131 (ELE)	1.43E-04	2.64E-04	1.55E-04	1.23E-04	1.14E-04	1.52E-04	1.27E-04
I-132	6.63E-07	1.09E-06	7.30E-07	5.85E-07	5.46E-07	6.46E-07	5.90E-07
Cs-137+D	1.04E-04	1.87E-04	1.14E-04	9.08E-05	8.51E-05	1.03E-04	9.29E-05
Ba-140+D	6.18E-03	1.07E-02	6.73E-03	5.46E-03	5.16E-03	5.95E-03	5.56E-03
Ce-141	2.18E-04	6.71E-04	2.43E-04	1.75E-04	1.59E-04	2.74E-04	1.59E-04
Ce-144+D	5.63E-05	2.12E-03	6.16E-05	4.81E-05	4.50E-05	5.98E-05	4.65E-05
Pr-143	6.39E-12	2.86E-04	7.06E-12	5.61E-12	5.25E-12	6.26E-12	5.71E-12
Pr-144	3.77E-05	2.08E-03	4.07E-05	3.35E-05	3.18E-05	3.58E-05	3.41E-05
TOTALS	3.61E-02	6.95E-02	3.99E-02	3.18E-02	2.98E-02	3.53E-02	3.23E-02

Table C-36. (continued).

Report Date: 18-May-91
Infant receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	6.25E-02	0.00E+00	0.00E+00	5.24E-01	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.11E-02	9.29E-04	9.29E-04	2.14E-03	9.29E-04	8.84E-02	6.33E-02
Y-91	8.97E-02	0.00E+00	0.00E+00	7.55E-01	0.00E+00	0.00E+00	0.00E+00
Zr-95	5.91E-01	4.50E-02	8.88E-02	3.77E+00	1.54E-01	2.95E-01	1.16E-01
Ru-103	2.46E-02	1.46E-03	2.46E-03	1.57E-01	3.94E-03	2.69E-03	2.24E-03
Ru-106	4.49E-02	6.86E-04	7.78E-04	3.57E-01	9.15E-04	9.15E-04	1.09E-03
I-131 (ELE)	2.66E-02	1.16E-04	8.87E-01	2.70E-03	1.27E-04	1.70E-04	1.74E-04
I-132	2.86E-07	2.26E-08	5.15E-06	8.04E-07	4.01E-08	3.43E-08	3.43E-08
Cs-137	2.39E-03	1.82E-03	2.21E-03	3.68E-03	2.21E-03	2.39E-03	3.13E-03
Ba-140	2.05E-03	0.00E+00	0.00E+00	3.58E-03	0.00E+00	5.06E-03	2.73E-03
Ce-141	6.66E-03	0.00E+00	0.00E+00	4.94E-02	0.00E+00	0.00E+00	0.00E+00
Ce-144	8.54E-01	2.81E-03	3.36E-03	6.70E+00	5.97E-02	1.04E-01	1.19E-01
Pr-143	1.40E-03	0.00E+00	0.00E+00	9.21E-03	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.23E-05	0.00E+00	0.00E+00	1.02E-04	0.00E+00	0.00E+00	0.00E+00
TOTALS	1.72E+00	5.28E-02	9.86E-01	1.23E+01	2.21E-01	4.99E-01	3.08E-01

Table C-36. (continued).

Report Date: 26-Jun-91
Infant receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	1.31E-04	5.69E+00	1.46E-04	1.16E-04	1.09E-04	1.27E-04	1.18E-04
Sr-90+D	0.00E+00	2.22E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	4.47E-03	8.29E+00	4.97E-03	3.97E-03	3.74E-03	4.25E-03	4.01E-03
Zr-95+D	5.66E+00	7.77E+00	6.27E+00	4.98E+00	4.66E+00	5.55E+00	5.07E+00
Ru-103+D	6.75E-02	9.92E-02	7.30E-02	5.87E-02	5.46E-02	6.88E-02	6.00E-02
Ru-106+D	1.08E-02	8.04E-01	1.18E-02	9.48E-03	8.86E-03	1.09E-02	9.68E-03
I-131 (ELE)	3.83E-02	1.01E-01	4.16E-02	3.29E-02	3.05E-02	4.06E-02	3.39E-02
I-132	1.92E-06	6.36E-06	2.12E-06	1.69E-06	1.58E-06	1.88E-06	1.72E-06
Cs-137+D	2.45E+00	7.78E+00	2.69E+00	2.15E+00	2.00E+00	2.43E+00	2.19E+00
Ba-140+D	2.18E-01	9.21E-01	2.38E-01	1.93E-01	1.82E-01	2.11E-01	1.95E-01
Ce-141	2.53E-02	7.92E-02	2.82E-02	2.02E-02	1.84E-02	3.17E-02	1.82E-02
Ce-144+D	4.80E-02	1.22E+01	5.27E-02	4.04E-02	3.75E-02	5.19E-02	3.80E-02
Pr-143	2.67E-10	6.68E-02	2.94E-10	2.35E-10	2.20E-10	2.62E-10	2.39E-10
Pr-144	1.19E-06	5.24E-04	1.29E-06	1.06E-06	1.00E-06	1.14E-06	1.08E-06
TOTALS	8.52E+00	6.60E+01	9.42E+00	7.49E+00	7.00E+00	8.39E+00	7.61E+00

Table C-36. (continued).

Report Date: 18-May-91
Infant receptor

9/18/58 Episodic Release
FPFRT-8

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	1.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	9.54E-01	8.07E-02	8.07E-02	8.07E-02	8.07E-02	7.34E+00	5.22E+00
Y-91	1.88E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.60E-01	1.05E-02	6.09E-03	1.05E-02	1.76E-02	3.67E-01	1.01E-01
Ru-103	5.46E-03	6.95E-04	7.80E-04	7.80E-04	1.06E-03	9.92E-04	1.13E-03
Ru-106	7.23E-03	1.79E-03	1.87E-03	1.87E-03	1.87E-03	2.20E-03	2.92E-03
I-131 (ELE)	5.28E+00	2.26E-02	1.78E+02	4.27E-02	2.21E-02	3.17E-02	3.36E-02
I-132	6.50E-12	5.41E-13	1.48E-10	1.08E-12	8.66E-13	7.58E-13	8.66E-13
Cs-137	2.74E-01	2.19E-01	2.60E-01	2.46E-01	2.60E-01	2.87E-01	3.69E-01
Ba-140	3.48E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	7.18E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	7.54E-02	1.51E-03	1.51E-03	1.51E-03	2.26E-02	6.49E-02	8.30E-02
Pr-143	1.37E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	5.19E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	6.88E+00	3.37E-01	1.78E+02	3.84E-01	4.06E-01	8.09E+00	5.81E+00

Table C-37. Dose evaluation for IET #3 (adult)

Report Date: 24-Jun-91
File Name: ACC8A.WK1

Episodic Release

Episode Description..... IET #3
Date..... 2/11-24/56
Receptor Location..... Building
Receptor Designation..... Adult
Release Location..... IET

Input Parameters

1.32E-10 : Episodic Mesoscale Dispersion Factor (hr/m**3)
3.46E+01 : Exposure Period (hr)
9.59E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 24-Jun-91

2/11-24/56 Episodic Release
IET #3

Summary of Dose Equivalents to Adult From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	5.71E+00	1.26E+01	6.22E+00	5.06E+00	4.78E+00	5.49E+00	5.13E+00
Inhalation	4.84E+00	0.00E+00	9.07E+01	1.40E+01	6.30E-02	1.85E+00	2.96E-01
Grnd Surface	1.13E+01	6.04E+01	1.24E+01	9.89E+00	9.25E+00	1.12E+01	1.01E+01
Ingestion	6.98E+00	0.00E+00	2.13E+02	4.17E-02	0.00E+00	1.90E+00	8.05E-01
TOTALS	2.88E+01	7.30E+01	3.22E+02	2.90E+01	1.41E+01	2.05E+01	1.63E+01

Table C-37. (continued).

Report Date: 24-Jun-91
Adult receptor

2/11-24/56 Episodic Release
IET #3

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Ar-41	1.83E+00	2.25E+03	8.58E-09	2.97E-07	0.00E+00
Br-84	5.30E-01	7.50E+02	2.86E-09	9.90E-08	1.80E-03
Kr-85m	4.48E+00	8.49E+02	3.24E-09	1.12E-07	0.00E+00
Kr-87	1.27E+00	4.28E+03	1.63E-08	5.65E-07	0.00E+00
Kr-88	2.84E+00	4.67E+03	1.78E-08	6.16E-07	0.00E+00
Rb-89	2.53E-01	1.63E+03	6.22E-09	2.15E-07	1.80E-03
Sr-89	1.21E+03	3.37E+01	1.29E-10	4.45E-09	1.80E-03
Sr-90	2.50E+05	2.28E-01	8.70E-13	3.01E-11	1.80E-03
Sr-91	9.50E+00	3.69E+02	1.41E-09	4.87E-08	1.80E-03
Sr-92	2.71E+00	7.90E+02	3.01E-09	1.04E-07	1.80E-03
Y-91	1.40E+03	3.38E+01	1.29E-10	4.46E-09	1.80E-03
Y-92	3.54E+00	3.48E+02	1.33E-09	4.59E-08	1.80E-03
Y-93	1.02E+01	3.84E+02	1.46E-09	5.07E-08	1.80E-03
Zr-95	1.54E+03	3.66E+01	1.40E-10	4.83E-09	1.80E-03
Zr-97	1.69E+01	2.83E+02	1.08E-09	3.74E-08	1.80E-03
Nb-96	2.34E+01	2.51E-02	9.58E-14	3.31E-12	1.80E-03
Mo-99	6.60E+01	1.79E+02	6.83E-10	2.36E-08	1.80E-03
Ru-103	9.44E+02	2.72E+01	1.04E-10	3.59E-09	1.80E-03
Ru-105	4.44E+00	1.06E+02	4.04E-10	1.40E-08	1.80E-03
Ru-106	8.84E+03	4.52E-01	1.72E-12	5.97E-11	1.80E-03
Sb-129	4.32E+00	7.04E+01	2.69E-10	9.29E-09	1.80E-03
Te-131	4.20E-01	5.06E+02	1.93E-09	6.68E-08	1.80E-03
Te-131m	3.00E+01	1.40E+01	5.34E-11	1.85E-09	1.80E-03
Te-132	7.80E+01	1.25E+02	4.77E-10	1.65E-08	1.80E-03
Te-133m	9.20E-01	4.98E+02	1.90E-09	6.57E-08	1.80E-03
Te-134	7.00E-01	1.05E+03	4.01E-09	1.39E-07	1.80E-03
I-131 (ELE)	1.93E+02	3.21E+02	1.22E-09	4.24E-08	1.80E-02
I-132	2.30E+00	5.19E+02	1.98E-09	6.85E-08	1.80E-02
I-133	2.08E+01	1.33E+03	5.07E-09	1.76E-07	1.80E-02
I-134	8.77E-01	5.55E+03	2.12E-08	7.33E-07	1.80E-02
I-135	6.61E+00	2.45E+03	9.35E-09	3.23E-07	1.80E-02
Xe-129m	2.13E+02	1.30E-06	4.96E-18	1.72E-16	0.00E+00
Xe-135	9.08E+00	1.78E+03	6.79E-09	2.35E-07	0.00E+00
Xe-135m	2.56E-01	7.42E+02	2.83E-09	9.79E-08	0.00E+00
Xe-138	2.36E-01	5.79E+03	2.21E-08	7.64E-07	0.00E+00
Cs-137	2.65E+05	2.34E-01	8.93E-13	3.09E-11	1.80E-03
Cs-138	5.37E-01	3.89E+03	1.48E-08	5.13E-07	1.80E-03
Ba-139	1.38E+00	1.22E+03	4.65E-09	1.61E-07	1.80E-03
Ba-140	3.07E+02	1.19E+02	4.54E-10	1.57E-08	1.80E-03
Ba-141	3.00E-01	6.82E+02	2.60E-09	9.00E-08	1.80E-03
Ba-142	1.80E-01	4.05E+02	1.55E-09	5.35E-08	1.80E-03
La-141	3.93E+00	6.38E+02	2.43E-09	8.42E-08	1.80E-03
La-142	1.59E+00	1.02E+03	3.89E-09	1.35E-07	1.80E-03
Ce-141	7.80E+02	5.63E+01	2.15E-10	7.43E-09	1.80E-03
Ce-143	3.30E+01	2.22E+02	8.47E-10	2.93E-08	1.80E-03
Ce-144	6.82E+03	7.57E+00	2.89E-11	9.99E-10	1.80E-03
Pr-143	3.25E+02	9.56E+01	3.65E-10	1.26E-08	1.80E-03
Pr-144	2.90E-01	7.55E+00	2.88E-11	9.97E-10	1.80E-03
U-234	2.14E+09	1.14E-02	4.35E-14	1.50E-12	1.80E-03
U-235	6.17E+12	3.62E-04	1.38E-15	4.78E-14	1.80E-03
U-238	3.91E+13	3.36E-06	1.28E-17	4.44E-16	1.80E-03
TOTAL		4.61E+04			

Table C-37. (continued).

Report Date: 24-Jun-91
Adult receptor

2/11-24/56 Episodic Release
IET #3

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	2.24E-01	4.13E-01	2.49E-01	1.99E-01	1.89E-01	2.13E-01	2.01E-01
Br-84	1.30E-01	3.24E-01	1.30E-01	1.64E-01	1.03E-01	1.14E-01	1.10E-01
Kr-85m	1.04E-02	3.35E-02	1.15E-02	8.60E-03	7.87E-03	1.26E-02	8.25E-03
Kr-87	2.88E-01	1.06E+00	3.07E-01	2.54E-01	2.42E-01	2.76E-01	2.60E-01
Kr-88	1.08E+00	2.94E+00	1.15E+00	9.62E-01	9.15E-01	1.02E+00	9.79E-01
Rb-89	2.95E-01	6.20E-01	3.13E-01	3.06E-01	2.41E-01	2.72E-01	2.60E-01
Sr-89	2.91E-01	2.37E+00	6.12E-05	1.32E+00	4.58E-05	5.31E-05	4.94E-05
Sr-90	1.78E-01	4.22E+00	0.00E+00	3.17E-01	0.00E+00	1.72E+00	7.55E-01
Sr-91	1.30E-01	3.47E-01	7.13E-02	4.21E-01	5.31E-02	6.34E-02	5.77E-02
Sr-92	1.84E-01	1.72E-01	1.19E-01	5.06E-01	9.03E-02	1.02E-01	9.65E-02
Y-91	1.92E-01	2.87E+00	1.73E-03	1.58E+00	1.30E-03	1.48E-03	1.40E-03
Y-92	3.72E-02	2.65E-01	1.09E-02	2.03E-01	8.15E-03	9.49E-03	8.81E-03
Y-93	1.08E-01	5.52E-01	6.63E-03	4.57E-01	5.05E-03	5.99E-03	5.44E-03
Zr-95	1.50E+00	1.87E+00	1.51E+00	1.20E+00	1.12E+00	3.05E+00	1.44E+00
Zr-97	2.55E-01	7.49E-01	1.22E-01	6.35E-01	9.11E-02	1.09E-01	9.91E-02
Nb-96	2.57E-05	3.40E-05	2.10E-05	3.55E-05	1.57E-05	1.86E-05	1.70E-05
Mo-99	1.06E-01	4.24E-01	2.30E-02	3.81E-01	1.69E-02	2.10E-02	1.82E-02
Ru-103	1.75E-01	1.88E-01	1.38E-01	3.14E-01	1.03E-01	1.30E-01	1.14E-01
Ru-105	1.57E-02	3.42E-02	1.13E-02	3.71E-02	8.36E-03	1.03E-02	9.12E-03
Ru-106	3.81E-02	6.14E-01	9.02E-03	2.19E-01	6.77E-03	8.31E-03	7.40E-03
Sb-129	1.77E-02	4.95E-02	1.40E-02	4.06E-02	1.05E-02	1.24E-02	1.14E-02
Te-131	2.37E-02	7.81E-02	1.54E-01	3.42E-02	1.39E-02	1.83E-02	1.49E-02
Te-131m	1.80E-02	2.00E-02	2.24E-01	2.13E-02	6.50E-03	7.83E-03	7.00E-03
Te-132	4.02E-01	8.22E-01	4.08E+00	3.40E-01	2.25E-01	2.73E-01	2.44E-01
Te-133m	1.28E-01	2.26E-01	7.20E-01	1.41E-01	8.61E-02	1.02E-01	9.28E-02
Te-134	9.00E-02	1.11E-01	3.50E-01	9.56E-02	6.18E-02	7.92E-02	6.72E-02
I-131 (ELE)	9.81E+00	6.49E+00	2.52E+02	2.13E+00	1.97E+00	2.62E+00	2.19E+00
I-132	3.84E-01	1.04E+00	8.14E-01	3.84E-01	2.99E-01	3.54E-01	3.24E-01
I-133	2.88E+00	1.21E+01	4.01E+01	1.52E+00	1.42E+00	1.75E+00	1.56E+00
I-134	2.47E+00	6.49E+00	3.42E+00	2.48E+00	1.98E+00	2.32E+00	2.13E+00
I-135	2.96E+00	8.40E+00	1.26E+01	2.83E+00	2.19E+00	2.49E+00	2.35E+00
Xe-129m	2.29E-12	1.88E-11	2.35E-12	1.43E-12	1.22E-12	2.25E-12	1.01E-12
Xe-135	3.35E-02	9.75E-02	3.64E-02	2.81E-02	2.60E-02	3.75E-02	2.87E-02
Xe-135m	2.40E-02	3.99E-02	2.60E-02	2.09E-02	1.94E-02	2.43E-02	2.13E-02
Xe-138	5.46E-01	1.12E+00	5.88E-01	4.84E-01	4.59E-01	5.27E-01	4.91E-01
Cs-137	2.85E-01	7.64E-01	3.07E-01	2.54E-01	1.97E-01	2.81E-01	2.58E-01
Cs-138	8.31E-01	1.89E+00	8.62E-01	9.91E-01	6.64E-01	7.50E-01	7.13E-01
Ba-139	2.87E-02	2.90E-01	4.36E-03	1.47E-01	3.00E-03	4.67E-03	3.13E-03
Ba-140	1.19E+00	4.08E+00	1.07E+00	9.62E-01	8.20E-01	1.08E+00	9.49E-01
Ba-141	5.53E-02	1.49E-01	5.36E-02	8.06E-02	3.98E-02	5.01E-02	4.33E-02
Ba-142	3.09E-02	6.15E-02	3.23E-02	3.58E-02	2.40E-02	2.87E-02	2.58E-02
La-141	4.67E-02	3.23E-01	3.38E-03	1.97E-01	6.56E-02	2.91E-03	2.76E-03
La-142	3.08E-01	6.06E-01	2.98E-01	3.92E-01	2.37E-01	2.62E-01	2.53E-01
Ce-141	1.31E-01	1.20E-01	4.28E-02	4.80E-01	2.78E-02	4.80E-02	2.75E-02
Ce-143	1.16E-01	2.98E-01	2.73E-02	4.14E-01	1.94E-02	2.65E-02	2.07E-02
Ce-144	4.14E-01	6.59E+00	2.85E-02	2.80E+00	2.03E-02	2.81E-02	2.06E-02
Pr-143	1.50E-01	6.99E-01	3.14E-09	5.81E-01	2.34E-09	2.79E-09	2.54E-09
Pr-144	6.11E-05	1.40E-03	2.26E-05	3.53E-04	1.77E-05	1.99E-05	1.90E-05
U-234	1.96E-01	2.17E-04	7.87E-06	1.59E+00	3.18E-06	1.28E-01	8.43E-03
U-235	5.96E-03	3.28E-04	2.31E-04	4.60E-02	1.54E-04	3.92E-03	4.06E-04
U-238	5.38E-05	2.56E-04	6.94E-07	4.26E-04	4.68E-07	3.47E-05	2.75E-06

Table C-37. (continued).

Report Date: 24-Jun-91
Adult receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	2.24E-01	4.13E-01	2.49E-01	1.99E-01	1.89E-01	2.13E-01	2.01E-01
Br-84	1.15E-01	2.64E-01	1.23E-01	1.02E-01	9.75E-02	1.08E-01	1.05E-01
Kr-85m	1.04E-02	3.35E-02	1.15E-02	8.60E-03	7.87E-03	1.26E-02	8.25E-03
Kr-87	2.88E-01	1.06E+00	3.07E-01	2.54E-01	2.42E-01	2.76E-01	2.60E-01
Kr-88+D	1.08E+00	2.94E+00	1.15E+00	9.62E-01	9.15E-01	1.02E+00	9.79E-01
Rb-89	2.80E-01	5.67E-01	3.04E-01	2.48E-01	2.35E-01	2.65E-01	2.53E-01
Sr-89	3.64E-07	2.22E-03	4.06E-07	3.22E-07	3.02E-07	3.52E-07	3.26E-07
Sr-90+D	0.00E+00	2.91E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	2.93E-02	6.75E-02	3.24E-02	2.58E-02	2.41E-02	2.88E-02	2.62E-02
Sr-92	8.24E-02	1.24E-01	9.08E-02	7.32E-02	6.92E-02	7.82E-02	7.39E-02
Y-91	9.51E-06	2.32E-03	1.06E-05	8.44E-06	7.99E-06	9.05E-06	8.54E-06
Y-92	6.91E-03	7.07E-02	7.65E-03	6.07E-03	5.71E-03	6.65E-03	6.18E-03
Y-93	2.79E-03	5.84E-02	3.03E-03	2.46E-03	2.32E-03	2.73E-03	2.50E-03
Zr-95+D	6.89E-03	9.60E-03	7.64E-03	6.07E-03	5.69E-03	6.75E-03	6.17E-03
Zr-97+D	3.50E-02	8.53E-02	3.85E-02	3.07E-02	2.88E-02	3.44E-02	3.13E-02
Nb-96	4.83E-06	7.10E-06	5.36E-06	4.27E-06	4.00E-06	4.72E-06	4.31E-06
Mo-99+D	2.15E-03	1.08E-02	2.38E-03	1.88E-03	1.75E-03	2.18E-03	1.90E-03
Ru-103+D	9.91E-04	1.39E-03	1.07E-03	8.64E-04	8.02E-04	1.01E-03	8.84E-04
Ru-105+D	6.47E-03	1.32E-02	7.13E-03	5.66E-03	5.29E-03	6.50E-03	5.77E-03
Ru-106+D	7.14E-06	8.71E-05	7.82E-06	6.27E-06	5.86E-06	7.14E-06	6.40E-06
Sb-129+D	8.27E-03	1.82E-02	9.14E-03	7.30E-03	6.85E-03	8.05E-03	7.40E-03
Te-131	1.64E-02	6.21E-02	1.81E-02	1.42E-02	1.32E-02	1.74E-02	1.41E-02
Te-131m+D	1.67E-03	2.68E-03	1.85E-03	1.47E-03	1.38E-03	1.65E-03	1.48E-03
Te-132+D	2.51E-02	4.08E-02	2.76E-02	2.20E-02	2.06E-02	2.48E-02	2.22E-02
Te-133m+D	9.37E-02	1.73E-01	1.03E-01	8.25E-02	7.74E-02	9.15E-02	8.33E-02
Te-134	7.02E-02	1.01E-01	7.71E-02	6.07E-02	5.66E-02	7.24E-02	6.15E-02
I-131 (ELE)	9.23E-03	1.70E-02	1.00E-02	7.92E-03	7.34E-03	9.76E-03	8.16E-03
I-132	9.29E-02	1.52E-01	1.02E-01	8.20E-02	7.66E-02	9.06E-02	8.28E-02
I-133	6.14E-02	1.39E-01	6.70E-02	5.38E-02	5.02E-02	6.16E-02	5.48E-02
I-134	1.15E+00	1.93E+00	1.28E+00	1.02E+00	9.52E-01	1.12E+00	1.03E+00
I-135	3.07E-01	5.05E-01	3.38E-01	2.73E-01	2.58E-01	2.93E-01	2.76E-01
Xe-129m	2.29E-12	1.88E-11	2.35E-12	1.43E-12	1.22E-12	2.25E-12	1.01E-12
Xe-135	3.35E-02	9.75E-02	3.64E-02	2.81E-02	2.60E-02	3.75E-02	2.87E-02
Xe-135m	2.40E-02	3.99E-02	2.60E-02	2.09E-02	1.94E-02	2.43E-02	2.13E-02
Xe-138	5.46E-01	1.12E+00	5.88E-01	4.84E-01	4.59E-01	5.27E-01	4.91E-01
Cs-137+D	1.02E-05	1.84E-05	1.12E-05	8.93E-06	8.36E-06	1.01E-05	9.13E-06
Cs-138	7.43E-01	1.56E+00	8.14E-01	6.61E-01	6.26E-01	7.08E-01	6.73E-01
Ba-139	3.30E-03	1.32E-01	3.65E-03	2.74E-03	2.52E-03	3.91E-03	2.63E-03
Ba-140+D	2.70E-02	4.70E-02	2.94E-02	2.39E-02	2.26E-02	2.60E-02	2.43E-02
Ba-141	4.71E-02	1.29E-01	5.16E-02	4.11E-02	3.84E-02	4.82E-02	4.17E-02
Ba-142	2.85E-02	5.78E-02	3.16E-02	2.50E-02	2.35E-02	2.81E-02	2.52E-02
La-141	2.14E-03	7.39E-02	2.35E-03	1.90E-03	1.80E-03	2.03E-03	1.92E-03
La-142	2.41E-01	4.14E-01	2.56E-01	2.15E-01	2.04E-01	2.26E-01	2.18E-01
Ce-141	3.27E-04	1.01E-03	3.65E-04	2.63E-04	2.39E-04	4.11E-04	2.39E-04
Ce-143	4.41E-03	1.56E-02	4.81E-03	3.71E-03	3.44E-03	4.64E-03	3.71E-03
Ce-144+D	3.05E-05	1.15E-03	3.33E-05	2.60E-05	2.43E-05	3.23E-05	2.51E-05
Pr-143	6.65E-11	2.98E-03	7.35E-11	5.84E-11	5.47E-11	6.52E-11	5.94E-11
Pr-144	2.03E-05	1.12E-03	2.19E-05	1.81E-05	1.72E-05	1.93E-05	1.84E-05
U-234	1.31E-10	5.61E-10	1.04E-10	7.05E-11	5.94E-11	1.22E-10	4.73E-11
U-235+D	4.52E-09	6.89E-09	4.98E-09	3.67E-09	3.35E-09	5.51E-09	3.47E-09
U-238+D	5.01E-12	3.33E-10	5.65E-12	4.19E-12	3.87E-12	5.65E-12	3.75E-12
TOTALS	5.71E+00	1.26E+01	6.22E+00	5.06E+00	4.78E+00	5.49E+00	5.13E+00

Table C-37. (continued).

Report Date: 24-Jun-91
Adult receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	8.26E-03	0.00E+00	0.00E+00	5.60E-02	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	7.63E-03	0.00E+00	0.00E+00	5.16E-02	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.58E-01	0.00E+00	0.00E+00	1.32E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	3.75E-02	0.00E+00	0.00E+00	3.17E-01	0.00E+00	0.00E+00	0.00E+00
Sr-91	6.54E-02	0.00E+00	0.00E+00	3.64E-01	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.70E-02	0.00E+00	0.00E+00	4.10E-01	0.00E+00	0.00E+00	0.00E+00
Y-91	1.88E-01	0.00E+00	0.00E+00	1.58E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	2.73E-02	0.00E+00	0.00E+00	1.94E-01	0.00E+00	0.00E+00	0.00E+00
Y-93	1.02E-01	0.00E+00	0.00E+00	4.52E-01	0.00E+00	0.00E+00	0.00E+00
Zr-95	8.80E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.71E+00	2.22E-01
Zr-97	1.43E-01	0.00E+00	0.00E+00	5.37E-01	0.00E+00	0.00E+00	0.00E+00
Nb-96	6.35E-06	0.00E+00	0.00E+00	1.87E-05	0.00E+00	0.00E+00	0.00E+00
Mo-99	8.16E-02	0.00E+00	0.00E+00	3.63E-01	0.00E+00	0.00E+00	0.00E+00
Ru-103	2.69E-02	0.00E+00	0.00E+00	2.03E-01	0.00E+00	0.00E+00	0.00E+00
Ru-105	5.50E-03	0.00E+00	0.00E+00	2.82E-02	0.00E+00	0.00E+00	0.00E+00
Ru-106	2.52E-02	0.00E+00	0.00E+00	2.12E-01	0.00E+00	0.00E+00	0.00E+00
Sb-129	5.08E-03	0.00E+00	0.00E+00	2.94E-02	0.00E+00	0.00E+00	0.00E+00
Te-131	6.34E-03	0.00E+00	1.35E-01	1.92E-02	0.00E+00	0.00E+00	0.00E+00
Te-131m	9.75E-03	0.00E+00	2.13E-01	1.44E-02	0.00E+00	0.00E+00	0.00E+00
Te-132	1.22E-01	0.00E+00	3.64E+00	9.97E-02	0.00E+00	0.00E+00	0.00E+00
Te-133m	2.40E-02	0.00E+00	6.05E-01	4.92E-02	0.00E+00	0.00E+00	0.00E+00
Te-134	1.33E-02	0.00E+00	2.66E-01	2.92E-02	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	1.30E+00	0.00E+00	4.47E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	2.17E-02	0.00E+00	4.14E-01	6.57E-02	0.00E+00	0.00E+00	0.00E+00
I-133	9.09E-01	0.00E+00	3.03E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	7.73E-02	0.00E+00	7.73E-01	3.65E-01	0.00E+00	0.00E+00	0.00E+00
I-135	3.41E-01	0.00E+00	9.61E+00	4.96E-01	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	9.48E-04	0.00E+00	8.59E-04	9.78E-04	0.00E+00	8.59E-04	9.18E-04
Cs-138	4.33E-02	0.00E+00	0.00E+00	2.91E-01	0.00E+00	0.00E+00	0.00E+00
Ba-139	2.47E-02	0.00E+00	0.00E+00	1.44E-01	0.00E+00	0.00E+00	0.00E+00
Ba-140	5.42E-02	0.00E+00	0.00E+00	9.49E-02	0.00E+00	1.34E-01	7.23E-02
Ba-141	6.39E-03	0.00E+00	0.00E+00	3.80E-02	0.00E+00	0.00E+00	0.00E+00
Ba-142	1.85E-03	0.00E+00	0.00E+00	1.03E-02	0.00E+00	0.00E+00	0.00E+00
La-141	4.36E-02	0.00E+00	0.00E+00	1.94E-01	6.30E-02	0.00E+00	0.00E+00
La-142	2.84E-02	0.00E+00	0.00E+00	1.42E-01	0.00E+00	0.00E+00	0.00E+00
Ce-141	6.06E-02	0.00E+00	0.00E+00	4.49E-01	0.00E+00	0.00E+00	0.00E+00
Ce-143	8.99E-02	0.00E+00	0.00E+00	3.93E-01	0.00E+00	0.00E+00	0.00E+00
Ce-144	3.35E-01	0.00E+00	0.00E+00	2.78E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	8.83E-02	0.00E+00	0.00E+00	5.81E-01	0.00E+00	0.00E+00	0.00E+00
Pr-144	4.01E-05	0.00E+00	0.00E+00	3.35E-04	0.00E+00	0.00E+00	0.00E+00
U-234	1.88E-01	0.00E+00	0.00E+00	1.59E+00	0.00E+00	0.00E+00	0.00E+00
U-235	5.50E-03	0.00E+00	0.00E+00	4.58E-02	0.00E+00	0.00E+00	0.00E+00
U-238	5.10E-05	0.00E+00	0.00E+00	4.25E-04	0.00E+00	0.00E+00	0.00E+00
TOTALS	4.84E+00	0.00E+00	9.07E+01	1.40E+01	6.30E-02	1.85E+00	2.96E-01

Table C-37. (continued).

Report Date: 24-Jun-91
Adult receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	6.23E-03	5.99E-02	6.70E-03	5.52E-03	5.25E-03	5.87E-03	5.64E-03
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	7.73E-03	5.37E-02	8.46E-03	6.87E-03	6.51E-03	7.36E-03	6.99E-03
Sr-89	5.48E-05	2.37E+00	6.08E-05	4.83E-05	4.55E-05	5.27E-05	4.91E-05
Sr-90+D	0.00E+00	4.22E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	3.52E-02	2.79E-01	3.89E-02	3.10E-02	2.90E-02	3.47E-02	3.15E-02
Sr-92	2.51E-02	4.81E-02	2.78E-02	2.24E-02	2.11E-02	2.38E-02	2.26E-02
Y-91	1.55E-03	2.87E+00	1.72E-03	1.37E-03	1.29E-03	1.47E-03	1.39E-03
Y-92	2.94E-03	1.94E-01	3.25E-03	2.60E-03	2.44E-03	2.84E-03	2.63E-03
Y-93	3.30E-03	4.94E-01	3.59E-03	2.90E-03	2.73E-03	3.26E-03	2.95E-03
Zr-95+D	1.35E+00	1.86E+00	1.50E+00	1.19E+00	1.12E+00	1.33E+00	1.21E+00
Zr-97+D	7.58E-02	6.64E-01	8.36E-02	6.66E-02	6.24E-02	7.48E-02	6.79E-02
Nb-96	1.42E-05	2.69E-05	1.57E-05	1.25E-05	1.17E-05	1.39E-05	1.27E-05
Mo-99+D	1.86E-02	4.13E-01	2.06E-02	1.62E-02	1.51E-02	1.89E-02	1.63E-02
Ru-103+D	1.27E-01	1.86E-01	1.37E-01	1.10E-01	1.03E-01	1.29E-01	1.13E-01
Ru-105+D	3.77E-03	2.10E-02	4.15E-03	3.29E-03	3.07E-03	3.80E-03	3.35E-03
Ru-106+D	8.26E-03	6.14E-01	9.01E-03	7.24E-03	6.77E-03	8.30E-03	7.40E-03
Sb-129+D	4.39E-03	3.13E-02	4.85E-03	3.88E-03	3.63E-03	4.30E-03	3.97E-03
Te-131	9.13E-04	1.61E-02	1.01E-03	7.83E-04	7.26E-04	9.71E-04	7.79E-04
Te-131m+D	6.24E-03	1.73E-02	6.90E-03	5.48E-03	5.12E-03	6.18E-03	5.52E-03
Te-132+D	2.50E-01	7.82E-01	2.76E-01	2.18E-01	2.05E-01	2.49E-01	2.22E-01
Te-133m+D	1.06E-02	5.22E-02	1.17E-02	9.32E-03	8.74E-03	1.04E-02	9.42E-03
Te-134	6.57E-03	9.12E-03	7.22E-03	5.68E-03	5.28E-03	6.81E-03	5.72E-03
I-131 (ELE)	2.47E+00	6.47E+00	2.67E+00	2.12E+00	1.96E+00	2.61E+00	2.18E+00
I-132	2.69E-01	8.91E-01	2.97E-01	2.36E-01	2.22E-01	2.63E-01	2.41E-01
I-133	1.68E+00	1.19E+01	1.83E+00	1.47E+00	1.37E+00	1.69E+00	1.50E+00
I-134	1.24E+00	4.56E+00	1.37E+00	1.09E+00	1.03E+00	1.20E+00	1.11E+00
I-135	2.31E+00	7.90E+00	2.54E+00	2.06E+00	1.93E+00	2.20E+00	2.07E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	2.41E-01	7.64E-01	2.65E-01	2.11E-01	1.97E-01	2.39E-01	2.15E-01
Cs-138	4.42E-02	3.29E-01	4.81E-02	3.93E-02	3.72E-02	4.22E-02	3.97E-02
Ba-139	6.42E-04	1.58E-01	7.10E-04	5.28E-04	4.84E-04	7.62E-04	5.03E-04
Ba-140+D	9.52E-01	4.03E+00	1.04E+00	8.43E-01	7.97E-01	9.22E-01	8.52E-01
Ba-141	1.80E-03	2.02E-02	1.97E-03	1.56E-03	1.46E-03	1.86E-03	1.59E-03
Ba-142	6.33E-04	3.75E-03	7.02E-04	5.55E-04	5.20E-04	6.28E-04	5.59E-04
La-141	9.36E-04	2.49E-01	1.03E-03	8.32E-04	7.88E-04	8.87E-04	8.40E-04
La-142	3.90E-02	1.92E-01	4.17E-02	3.47E-02	3.31E-02	3.67E-02	3.55E-02
Ce-141	3.80E-02	1.19E-01	4.24E-02	3.04E-02	2.76E-02	4.76E-02	2.73E-02
Ce-143	2.06E-02	2.83E-01	2.25E-02	1.72E-02	1.59E-02	2.19E-02	1.70E-02
Ce-144+D	2.59E-02	6.59E+00	2.85E-02	2.18E-02	2.03E-02	2.81E-02	2.06E-02
Pr-143	2.78E-09	6.96E-01	3.06E-09	2.44E-09	2.29E-09	2.73E-09	2.49E-09
Pr-144	6.43E-07	2.83E-04	6.94E-07	5.71E-07	5.41E-07	6.12E-07	5.80E-07
U-234	2.75E-05	2.17E-04	7.87E-06	5.93E-06	3.18E-06	1.01E-05	3.12E-06
U-235+D	2.14E-04	3.28E-04	2.31E-04	1.69E-04	1.54E-04	2.55E-04	1.59E-04
U-238+D	6.44E-07	2.56E-04	6.94E-07	5.11E-07	4.68E-07	7.13E-07	4.47E-07
TOTALS	1.13E+01	6.04E+01	1.24E+01	9.89E+00	9.25E+00	1.12E+01	1.01E+01

Table C-37. (continued).

Report Date: 24-Jun-91
Adult receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	1.30E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.58E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.33E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.40E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.72E+00	7.55E-01
Sr-91	1.72E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.91E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.45E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	1.90E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	2.95E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	4.82E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-97	1.23E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	3.83E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	3.99E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	2.04E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-105	9.89E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	4.62E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-129	1.07E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	1.75E-23	0.00E+00	2.89E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	3.08E-04	0.00E+00	2.88E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	4.55E-03	0.00E+00	1.35E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	4.72E-13	0.00E+00	9.95E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	5.92E-16	0.00E+00	9.31E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	6.04E+00	0.00E+00	2.05E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	4.42E-07	0.00E+00	1.09E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	2.31E-01	0.00E+00	7.86E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	4.80E-12	0.00E+00	5.81E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.81E-03	0.00E+00	9.41E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	4.35E-02	0.00E+00	4.17E-02	4.17E-02	0.00E+00	4.17E-02	4.17E-02
Cs-138	1.04E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	3.61E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.52E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	7.11E-30	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	2.04E-47	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	4.60E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	2.63E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	3.26E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	1.53E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	5.25E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	6.17E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.79E-33	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-234	8.11E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.28E-01	8.43E-03
U-235	2.48E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.67E-03	2.48E-04
U-238	2.12E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.40E-05	2.30E-06
TOTALS	6.98E+00	0.00E+00	2.13E+02	4.17E-02	0.00E+00	1.90E+00	8.05E-01

Table C-38. Dose evaluation for IET #3 (infant)

Report Date: 24-Jun-91
File Name: ACC8B.WK1

Episodic Release

Episode Description..... IET #3
Date..... 2/11-24/56
Receptor Location..... Building
Receptor Designation..... Infant
Release Location..... IET

Input Parameters

1.32E-10 : Episodic Mesoscale Dispersion Factor (hr/m**3)
3.46E+01 : Exposure Period (hr)
1.58E-01 : Breathing Rate (m**3/hr)
5.00E+01 : Ground Surface Deposition Dose Commitment Period (y) [4.38E+05 hours]
0.7 : Shielding Reduction Factor - Ground Surface Deposition

Report Date: 24-Jun-91

2/11-24/56 Episodic Release
IET #3

Summary of Dose Equivalents to Infant From All Pathways (mrem)

Pathway	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Immersion	5.71E+00	1.26E+01	6.22E+00	5.06E+00	4.78E+00	5.49E+00	5.13E+00
Inhalation	2.93E+00	2.65E-02	6.54E+01	7.02E+00	1.02E-01	1.88E-01	1.37E-01
Grnd Surface	1.13E+01	6.04E+01	1.24E+01	9.89E+00	9.25E+00	1.12E+01	1.01E+01
Ingestion	3.45E+01	1.88E-01	1.15E+03	3.20E-01	2.03E-01	1.77E+00	1.32E+00
TOTALS	5.44E+01	7.32E+01	1.23E+03	2.23E+01	1.43E+01	1.87E+01	1.66E+01

Table C-39. (continued).

Report Date: 24-Jun-91
Infant receptor

2/11-24/56 Episodic Release
IET #3

Source Term and Average Air Concentrations

Radionuclide	Half-life (hours)	Activity Released (Ci)	Average Air Concen. At Receptor (Ci/m3)	Total Integrated Air Concentration (Ci-h/m3)	Deposition Velocity (m/s)
Ar-41	1.83E+00	2.25E+03	8.58E-09	2.97E-07	0.00E+00
Br-84	5.30E-01	7.50E+02	2.86E-09	9.90E-08	1.80E-03
Kr-85m	4.48E+00	8.49E+02	3.24E-09	1.12E-07	0.00E+00
Kr-87	1.27E+00	4.28E+03	1.63E-08	5.65E-07	0.00E+00
Kr-88	2.84E+00	4.67E+03	1.78E-08	6.16E-07	0.00E+00
Rb-89	2.53E-01	1.63E+03	6.22E-09	2.15E-07	1.80E-03
Sr-89	1.21E+03	3.37E+01	1.29E-10	4.45E-09	1.80E-03
Sr-90	2.50E+05	2.28E-01	8.70E-13	3.01E-11	1.80E-03
Sr-91	9.50E+00	3.69E+02	1.41E-09	4.87E-08	1.80E-03
Sr-92	2.71E+00	7.90E+02	3.01E-09	1.04E-07	1.80E-03
Y-91	1.40E+03	3.38E+01	1.29E-10	4.46E-09	1.80E-03
Y-92	3.54E+00	3.48E+02	1.33E-09	4.59E-08	1.80E-03
Y-93	1.02E+01	3.84E+02	1.46E-09	5.07E-08	1.80E-03
Zr-95	1.54E+03	3.66E+01	1.40E-10	4.83E-09	1.80E-03
Zr-97	1.69E+01	2.83E+02	1.08E-09	3.74E-08	1.80E-03
Nb-96	2.34E+01	2.51E-02	9.58E-14	3.31E-12	1.80E-03
Mo-99	6.60E+01	1.79E+02	6.83E-10	2.36E-08	1.80E-03
Ru-103	9.44E+02	2.72E+01	1.04E-10	3.59E-09	1.80E-03
Ru-105	4.44E+00	1.06E+02	4.04E-10	1.40E-08	1.80E-03
Ru-106	8.84E+03	4.52E-01	1.72E-12	5.97E-11	1.80E-03
Sb-129	4.32E+00	7.04E+01	2.69E-10	9.29E-09	1.80E-03
Te-131	4.20E-01	5.06E+02	1.93E-09	6.68E-08	1.80E-03
Te-131m	3.00E+01	1.40E+01	5.34E-11	1.85E-09	1.80E-03
Te-132	7.80E+01	1.25E+02	4.77E-10	1.65E-08	1.80E-03
Te-133m	9.20E-01	4.98E+02	1.90E-09	6.57E-08	1.80E-03
Te-134	7.00E-01	1.05E+03	4.01E-09	1.39E-07	1.80E-03
I-131 (ELE)	1.93E+02	3.21E+02	1.22E-09	4.24E-08	1.80E-02
I-132	2.30E+00	5.19E+02	1.98E-09	6.85E-08	1.80E-02
I-133	2.08E+01	1.33E+03	5.07E-09	1.76E-07	1.80E-02
I-134	8.77E-01	5.55E+03	2.12E-08	7.33E-07	1.80E-02
I-135	6.61E+00	2.45E+03	9.35E-09	3.23E-07	1.80E-02
Xe-129m	2.13E+02	1.30E-06	4.96E-18	1.72E-16	0.00E+00
Xe-135	9.08E+00	1.78E+03	6.79E-09	2.35E-07	0.00E+00
Xe-135m	2.56E-01	7.42E+02	2.83E-09	9.79E-08	0.00E+00
Xe-138	2.36E-01	5.79E+03	2.21E-08	7.64E-07	0.00E+00
Cs-137	2.65E+05	2.34E-01	8.93E-13	3.09E-11	1.80E-03
Cs-138	5.37E-01	3.89E+03	1.48E-08	5.13E-07	1.80E-03
Ba-139	1.38E+00	1.22E+03	4.65E-09	1.61E-07	1.80E-03
Ba-140	3.07E+02	1.19E+02	4.54E-10	1.57E-08	1.80E-03
Ba-141	3.00E-01	6.82E+02	2.60E-09	9.00E-08	1.80E-03
Ba-142	1.80E-01	4.05E+02	1.55E-09	5.35E-08	1.80E-03
La-141	3.93E+00	6.38E+02	2.43E-09	8.42E-08	1.80E-03
La-142	1.59E+00	1.02E+03	3.89E-09	1.35E-07	1.80E-03
Ce-141	7.80E+02	5.63E+01	2.15E-10	7.43E-09	1.80E-03
Ce-143	3.30E+01	2.22E+02	8.47E-10	2.93E-08	1.80E-03
Ce-144	6.82E+03	7.57E+00	2.89E-11	9.99E-10	1.80E-03
Pr-143	3.25E+02	9.56E+01	3.65E-10	1.26E-08	1.80E-03
Pr-144	2.90E-01	7.55E+00	2.88E-11	9.97E-10	1.80E-03
U-234	2.14E+09	1.14E-02	4.35E-14	1.50E-12	1.80E-03
U-235	6.17E+12	3.62E-04	1.38E-15	4.78E-14	1.80E-03
U-238	3.91E+13	3.36E-06	1.28E-17	4.44E-16	1.80E-03
TOTAL		4.61E+04			

Table C-39. (continued).

Report Date: 24-Jun-91
Infant Receptor

2/11-24/56 Episodic Release
IET #3

Summary of Dose Equivalents by Radionuclide (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	2.24E-01	4.13E-01	2.49E-01	1.99E-01	1.89E-01	2.13E-01	2.01E-01
Br-84	1.23E-01	3.24E-01	1.30E-01	1.17E-01	1.03E-01	1.14E-01	1.10E-01
Kr-85m	1.04E-02	3.35E-02	1.15E-02	8.60E-03	7.87E-03	1.26E-02	8.25E-03
Kr-87	2.88E-01	1.06E+00	3.07E-01	2.54E-01	2.42E-01	2.76E-01	2.60E-01
Kr-88	1.08E+00	2.94E+00	1.15E+00	9.62E-01	9.15E-01	1.02E+00	9.79E-01
Rb-89	2.89E-01	6.20E-01	3.13E-01	2.63E-01	2.41E-01	2.72E-01	2.60E-01
Sr-89	6.81E-02	2.37E+00	6.12E-05	2.18E-01	4.58E-05	5.31E-05	4.94E-05
Sr-90	1.83E-01	4.23E+00	1.55E-02	1.57E-02	1.55E-02	1.41E+00	1.00E+00
Sr-91	7.53E-02	3.47E-01	7.13E-02	1.17E-01	5.31E-02	6.34E-02	5.77E-02
Sr-92	1.20E-01	1.72E-01	1.19E-01	1.63E-01	9.03E-02	1.02E-01	9.65E-02
Y-91	3.92E-02	2.87E+00	1.73E-03	2.63E-01	1.30E-03	1.48E-03	1.40E-03
Y-92	1.44E-02	2.65E-01	1.09E-02	4.07E-02	8.15E-03	9.49E-03	8.81E-03
Y-93	2.29E-02	5.52E-01	6.63E-03	8.00E-02	5.05E-03	5.99E-03	5.44E-03
Zr-95	1.54E+00	1.88E+00	1.53E+00	2.10E+00	1.16E+00	1.49E+00	1.27E+00
Zr-97	1.34E-01	7.49E-01	1.22E-01	1.06E-01	9.11E-02	1.09E-01	9.91E-02
Nb-96	2.04E-05	3.40E-05	2.10E-05	1.99E-05	1.57E-05	1.86E-05	1.70E-05
Mo-99	3.53E-02	4.24E-01	2.30E-02	7.79E-02	1.69E-02	2.10E-02	1.82E-02
Ru-103	1.84E-01	1.92E-01	1.44E-01	4.07E-01	1.13E-01	1.37E-01	1.20E-01
Ru-105	1.12E-02	3.42E-02	1.13E-02	1.36E-02	8.36E-03	1.03E-02	9.12E-03
Ru-106	4.81E-02	6.16E-01	1.10E-02	2.82E-01	8.90E-03	1.07E-02	1.05E-02
Sb-129	1.35E-02	4.95E-02	1.40E-02	1.60E-02	1.05E-02	1.24E-02	1.14E-02
Te-131	1.84E-02	7.81E-02	4.13E-02	1.81E-02	1.39E-02	1.83E-02	1.49E-02
Te-131m	9.53E-03	2.00E-02	4.40E-02	9.31E-03	6.50E-03	7.83E-03	7.00E-03
Te-132	2.96E-01	8.22E-01	9.13E-01	2.57E-01	2.25E-01	2.73E-01	2.44E-01
Te-133m	1.08E-01	2.26E-01	2.15E-01	9.99E-02	8.61E-02	1.02E-01	9.28E-02
Te-134	7.89E-02	1.11E-01	1.28E-01	7.12E-02	6.18E-02	7.92E-02	6.72E-02
I-131 (ELE)	3.81E+01	6.64E+00	1.20E+03	2.57E+00	2.12E+00	2.84E+00	2.41E+00
I-132	4.02E-01	1.05E+00	1.12E+00	4.31E-01	3.04E-01	3.59E-01	3.29E-01
I-133	2.03E+00	1.21E+01	1.15E+01	1.52E+00	1.42E+00	1.75E+00	1.56E+00
I-134	2.40E+00	6.49E+00	2.78E+00	2.17E+00	1.98E+00	2.32E+00	2.13E+00
I-135	2.68E+00	8.40E+00	4.48E+00	2.41E+00	2.19E+00	2.49E+00	2.35E+00
Xe-129m	2.29E-12	1.88E-11	2.35E-12	1.43E-12	1.22E-12	2.25E-12	1.01E-12
Xe-135	3.35E-02	9.75E-02	3.64E-02	2.81E-02	2.60E-02	3.75E-02	2.87E-02
Xe-135m	2.40E-02	3.99E-02	2.60E-02	2.09E-02	1.94E-02	2.43E-02	2.13E-02
Xe-138	5.46E-01	1.12E+00	5.88E-01	4.84E-01	4.59E-01	5.27E-01	4.91E-01
Cs-137	2.68E-01	7.86E-01	2.90E-01	2.36E-01	2.23E-01	2.67E-01	2.52E-01
Cs-138	7.95E-01	1.89E+00	8.62E-01	7.49E-01	6.64E-01	7.50E-01	7.13E-01
Ba-139	8.03E-03	2.90E-01	4.36E-03	2.70E-02	3.00E-03	4.67E-03	3.13E-03
Ba-140	1.00E+00	4.08E+00	1.07E+00	8.83E-01	8.20E-01	9.70E-01	8.88E-01
Ba-141	5.00E-02	1.49E-01	5.36E-02	4.89E-02	3.98E-02	5.01E-02	4.33E-02
Ba-142	2.94E-02	6.15E-02	3.23E-02	2.72E-02	2.40E-02	2.87E-02	2.58E-02
La-141	1.03E-02	3.23E-01	3.38E-03	3.47E-02	1.30E-02	2.91E-03	2.76E-03
La-142	2.85E-01	6.06E-01	2.98E-01	2.73E-01	2.37E-01	2.62E-01	2.53E-01
Ce-141	4.94E-02	1.20E-01	4.28E-02	1.05E-01	2.78E-02	4.80E-02	2.75E-02
Ce-143	3.98E-02	2.98E-01	2.73E-02	8.59E-02	1.94E-02	2.65E-02	2.07E-02
Ce-144	5.29E-01	6.59E+00	3.12E-02	3.64E+00	6.48E-02	1.19E-01	1.30E-01
Pr-143	1.60E-02	6.99E-01	3.14E-09	9.59E-02	2.34E-09	2.79E-09	2.54E-09
Pr-144	2.76E-05	1.40E-03	2.26E-05	7.39E-05	1.77E-05	1.99E-05	1.90E-05
U-234	3.24E-02	2.17E-04	7.87E-06	2.62E-01	3.18E-06	2.26E-02	1.49E-03
U-235	1.17E-03	3.28E-04	2.31E-04	7.73E-03	1.54E-04	9.02E-04	2.02E-04
U-238	9.44E-06	2.56E-04	6.94E-07	7.07E-05	4.68E-07	6.72E-06	8.52E-07

Table C-39. (continued).

Report Date: 24-Jun-91
Infant receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Air Immersion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	2.24E-01	4.13E-01	2.49E-01	1.99E-01	1.89E-01	2.13E-01	2.01E-01
Br-84	1.15E-01	2.64E-01	1.23E-01	1.02E-01	9.75E-02	1.08E-01	1.05E-01
Kr-85m	1.04E-02	3.35E-02	1.15E-02	8.60E-03	7.87E-03	1.26E-02	8.25E-03
Kr-87	2.88E-01	1.06E+00	3.07E-01	2.54E-01	2.42E-01	2.76E-01	2.60E-01
Kr-88+D	1.08E+00	2.94E+00	1.15E+00	9.62E-01	9.15E-01	1.02E+00	9.79E-01
Rb-89	2.80E-01	5.67E-01	3.04E-01	2.48E-01	2.35E-01	2.65E-01	2.53E-01
Sr-89	3.64E-07	2.22E-03	4.06E-07	3.22E-07	3.02E-07	3.52E-07	3.26E-07
Sr-90+D	0.00E+00	2.91E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	2.93E-02	6.75E-02	3.24E-02	2.58E-02	2.41E-02	2.88E-02	2.62E-02
Sr-92	8.24E-02	1.24E-01	9.08E-02	7.32E-02	6.92E-02	7.82E-02	7.39E-02
Y-91	9.51E-06	2.32E-03	1.06E-05	8.44E-06	7.99E-06	9.05E-06	8.54E-06
Y-92	6.91E-03	7.07E-02	7.65E-03	6.07E-03	5.71E-03	6.65E-03	6.18E-03
Y-93	2.79E-03	5.84E-02	3.03E-03	2.46E-03	2.32E-03	2.73E-03	2.50E-03
Zr-95+D	6.89E-03	9.60E-03	7.64E-03	6.07E-03	5.69E-03	6.75E-03	6.17E-03
Zr-97+D	3.50E-02	8.53E-02	3.85E-02	3.07E-02	2.88E-02	3.44E-02	3.13E-02
Nb-96	4.83E-06	7.10E-06	5.36E-06	4.27E-06	4.00E-06	4.72E-06	4.31E-06
Mo-99+D	2.15E-03	1.08E-02	2.38E-03	1.88E-03	1.75E-03	2.18E-03	1.90E-03
Ru-103+D	9.91E-04	1.39E-03	1.07E-03	8.64E-04	8.02E-04	1.01E-03	8.84E-04
Ru-105+D	6.47E-03	1.32E-02	7.13E-03	5.66E-03	5.29E-03	6.50E-03	5.77E-03
Ru-106+D	7.14E-06	8.71E-05	7.82E-06	6.27E-06	5.86E-06	7.14E-06	6.40E-06
Sb-129+D	8.27E-03	1.82E-02	9.14E-03	7.30E-03	6.85E-03	8.05E-03	7.40E-03
Te-131	1.64E-02	6.21E-02	1.81E-02	1.42E-02	1.32E-02	1.74E-02	1.41E-02
Te-131m+D	1.67E-03	2.68E-03	1.85E-03	1.47E-03	1.38E-03	1.65E-03	1.48E-03
Te-132+D	2.51E-02	4.08E-02	2.76E-02	2.20E-02	2.06E-02	2.48E-02	2.22E-02
Te-133m+D	9.37E-02	1.73E-01	1.03E-01	8.25E-02	7.74E-02	9.15E-02	8.33E-02
Te-134	7.02E-02	1.01E-01	7.71E-02	6.07E-02	5.66E-02	7.24E-02	6.15E-02
I-131 (ELE)	9.23E-03	1.70E-02	1.00E-02	7.92E-03	7.34E-03	9.76E-03	8.16E-03
I-132	9.29E-02	1.52E-01	1.02E-01	8.20E-02	7.66E-02	9.06E-02	8.28E-02
I-133	6.14E-02	1.39E-01	6.70E-02	5.38E-02	5.02E-02	6.16E-02	5.48E-02
I-134	1.15E+00	1.93E+00	1.28E+00	1.02E+00	9.52E-01	1.12E+00	1.03E+00
I-135	3.07E-01	5.05E-01	3.38E-01	2.73E-01	2.58E-01	2.93E-01	2.76E-01
Xe-129m	2.29E-12	1.88E-11	2.35E-12	1.43E-12	1.22E-12	2.25E-12	1.01E-12
Xe-135	3.35E-02	9.75E-02	3.64E-02	2.81E-02	2.60E-02	3.75E-02	2.87E-02
Xe-135m	2.40E-02	3.99E-02	2.60E-02	2.09E-02	1.94E-02	2.43E-02	2.13E-02
Xe-138	5.46E-01	1.12E+00	5.88E-01	4.84E-01	4.59E-01	5.27E-01	4.91E-01
Cs-137+D	1.02E-05	1.84E-05	1.12E-05	8.93E-06	8.36E-06	1.01E-05	9.13E-06
Cs-138	7.43E-01	1.56E+00	8.14E-01	6.61E-01	6.26E-01	7.08E-01	6.73E-01
Ba-139	3.30E-03	1.32E-01	3.65E-03	2.74E-03	2.52E-03	3.91E-03	2.63E-03
Ba-140+D	2.70E-02	4.70E-02	2.94E-02	2.39E-02	2.26E-02	2.60E-02	2.43E-02
Ba-141	4.71E-02	1.29E-01	5.16E-02	4.11E-02	3.84E-02	4.82E-02	4.17E-02
Ba-142	2.85E-02	5.78E-02	3.16E-02	2.50E-02	2.35E-02	2.81E-02	2.52E-02
La-141	2.14E-03	7.39E-02	2.35E-03	1.90E-03	1.80E-03	2.03E-03	1.92E-03
La-142	2.41E-01	4.14E-01	2.56E-01	2.15E-01	2.04E-01	2.26E-01	2.18E-01
Ce-141	3.27E-04	1.01E-03	3.65E-04	2.63E-04	2.39E-04	4.11E-04	2.39E-04
Ce-143	4.41E-03	1.56E-02	4.81E-03	3.71E-03	3.44E-03	4.64E-03	3.71E-03
Ce-144+D	3.05E-05	1.15E-03	3.33E-05	2.60E-05	2.43E-05	3.23E-05	2.51E-05
Pr-143	6.65E-11	2.98E-03	7.35E-11	5.84E-11	5.47E-11	6.52E-11	5.94E-11
Pr-144	2.03E-05	1.12E-03	2.19E-05	1.81E-05	1.72E-05	1.93E-05	1.84E-05
U-234	1.31E-10	5.61E-10	1.04E-10	7.05E-11	5.94E-11	1.22E-10	4.73E-11
U-235+D	4.52E-09	6.89E-09	4.98E-09	3.67E-09	3.35E-09	5.51E-09	3.47E-09
U-238+D	5.01E-12	3.33E-10	5.65E-12	4.19E-12	3.87E-12	5.65E-12	3.75E-12
TOTALS	5.71E+00	1.26E+01	6.22E+00	5.06E+00	4.78E+00	5.49E+00	5.13E+00

Table C-39. (continued).

Report Date: 24-Jun-91
Infant receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Inhalation (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	1.36E-03	0.00E+00	0.00E+00	9.25E-03	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	1.26E-03	0.00E+00	0.00E+00	8.51E-03	0.00E+00	0.00E+00	0.00E+00
Sr-89	2.61E-02	0.00E+00	0.00E+00	2.18E-01	0.00E+00	0.00E+00	0.00E+00
Sr-90	2.12E-03	1.76E-04	1.76E-04	4.05E-04	1.76E-04	1.68E-02	1.20E-02
Sr-91	1.08E-02	0.00E+00	0.00E+00	6.01E-02	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.27E-02	0.00E+00	0.00E+00	6.77E-02	0.00E+00	0.00E+00	0.00E+00
Y-91	3.11E-02	0.00E+00	0.00E+00	2.61E-01	0.00E+00	0.00E+00	0.00E+00
Y-92	4.51E-03	0.00E+00	0.00E+00	3.20E-02	0.00E+00	0.00E+00	0.00E+00
Y-93	1.69E-02	0.00E+00	0.00E+00	7.46E-02	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.41E-01	1.08E-02	2.13E-02	9.02E-01	3.68E-02	7.07E-02	2.78E-02
Zr-97	2.37E-02	0.00E+00	0.00E+00	8.87E-03	0.00E+00	0.00E+00	0.00E+00
Nb-96	1.05E-06	0.00E+00	0.00E+00	3.09E-06	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.35E-02	0.00E+00	0.00E+00	5.98E-02	0.00E+00	0.00E+00	0.00E+00
Ru-103	4.63E-02	2.73E-03	4.63E-03	2.94E-01	7.39E-03	5.05E-03	4.21E-03
Ru-105	9.08E-04	0.00E+00	0.00E+00	4.65E-03	0.00E+00	0.00E+00	0.00E+00
Ru-106	3.43E-02	5.24E-04	5.94E-04	2.73E-01	6.99E-04	6.99E-04	8.31E-04
Sb-129	8.38E-04	0.00E+00	0.00E+00	4.85E-03	0.00E+00	0.00E+00	0.00E+00
Te-131	1.05E-03	0.00E+00	2.22E-02	3.17E-03	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.61E-03	0.00E+00	3.51E-02	2.37E-03	0.00E+00	0.00E+00	0.00E+00
Te-132	2.01E-02	0.00E+00	6.01E-01	1.65E-02	0.00E+00	0.00E+00	0.00E+00
Te-133m	3.95E-03	0.00E+00	9.99E-02	8.12E-03	0.00E+00	0.00E+00	0.00E+00
Te-134	2.19E-03	0.00E+00	4.39E-02	4.83E-03	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	1.71E+00	7.45E-03	5.71E+01	1.74E-01	8.18E-03	1.09E-02	1.12E-02
I-132	4.01E-02	3.17E-03	7.22E-01	1.13E-01	5.62E-03	4.82E-03	4.82E-03
I-133	1.50E-01	0.00E+00	5.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.28E-02	0.00E+00	1.28E-01	6.03E-02	0.00E+00	0.00E+00	0.00E+00
I-135	5.63E-02	0.00E+00	1.59E+00	8.19E-02	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	2.35E-04	1.79E-04	2.17E-04	3.62E-04	2.17E-04	2.35E-04	3.08E-04
Cs-138	7.15E-03	0.00E+00	0.00E+00	4.80E-02	0.00E+00	0.00E+00	0.00E+00
Ba-139	4.08E-03	0.00E+00	0.00E+00	2.37E-02	0.00E+00	0.00E+00	0.00E+00
Ba-140	8.95E-03	0.00E+00	0.00E+00	1.57E-02	0.00E+00	2.21E-02	1.19E-02
Ba-141	1.05E-03	0.00E+00	0.00E+00	6.27E-03	0.00E+00	0.00E+00	0.00E+00
Ba-142	3.05E-04	0.00E+00	0.00E+00	1.69E-03	0.00E+00	0.00E+00	0.00E+00
La-141	7.20E-03	0.00E+00	0.00E+00	3.20E-02	1.04E-02	0.00E+00	0.00E+00
La-142	4.69E-03	0.00E+00	0.00E+00	2.34E-02	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.00E-02	0.00E+00	0.00E+00	7.41E-02	0.00E+00	0.00E+00	0.00E+00
Ce-143	1.48E-02	0.00E+00	0.00E+00	6.49E-02	0.00E+00	0.00E+00	0.00E+00
Ce-144	4.62E-01	1.52E-03	1.82E-03	3.62E+00	3.23E-02	5.62E-02	6.44E-02
Pr-143	1.46E-02	0.00E+00	0.00E+00	9.59E-02	0.00E+00	0.00E+00	0.00E+00
Pr-144	6.63E-06	0.00E+00	0.00E+00	5.52E-05	0.00E+00	0.00E+00	0.00E+00
U-234	3.10E-02	0.00E+00	0.00E+00	2.62E-01	0.00E+00	0.00E+00	0.00E+00
U-235	9.08E-04	0.00E+00	0.00E+00	7.56E-03	0.00E+00	0.00E+00	0.00E+00
U-238	8.43E-06	0.00E+00	0.00E+00	7.02E-05	0.00E+00	0.00E+00	0.00E+00
TOTALS	2.93E+00	2.65E-02	6.54E+01	7.02E+00	1.02E-01	1.88E-01	1.37E-01

Table C-39. (continued).

Report Date: 24-Jun-91
Infant receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Contaminated Ground Surface (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	6.23E-03	5.99E-02	6.70E-03	5.52E-03	5.25E-03	5.87E-03	5.64E-03
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88+D	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	7.73E-03	5.37E-02	8.46E-03	6.87E-03	6.51E-03	7.36E-03	6.99E-03
Sr-89	5.48E-05	2.37E+00	6.08E-05	4.83E-05	4.55E-05	5.27E-05	4.91E-05
Sr-90+D	0.00E+00	4.22E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91+D	3.52E-02	2.79E-01	3.89E-02	3.10E-02	2.90E-02	3.47E-02	3.15E-02
Sr-92	2.51E-02	4.81E-02	2.78E-02	2.24E-02	2.11E-02	2.38E-02	2.26E-02
Y-91	1.55E-03	2.87E+00	1.72E-03	1.37E-03	1.29E-03	1.47E-03	1.39E-03
Y-92	2.94E-03	1.94E-01	3.25E-03	2.60E-03	2.44E-03	2.84E-03	2.63E-03
Y-93	3.30E-03	4.94E-01	3.59E-03	2.90E-03	2.73E-03	3.26E-03	2.95E-03
Zr-95+D	1.35E+00	1.86E+00	1.50E+00	1.19E+00	1.12E+00	1.33E+00	1.21E+00
Zr-97+D	7.58E-02	6.64E-01	8.36E-02	6.66E-02	6.24E-02	7.48E-02	6.79E-02
Nb-96	1.42E-05	2.69E-05	1.57E-05	1.25E-05	1.17E-05	1.39E-05	1.27E-05
Mo-99+D	1.86E-02	4.13E-01	2.06E-02	1.62E-02	1.51E-02	1.89E-02	1.63E-02
Ru-103+D	1.27E-01	1.86E-01	1.37E-01	1.10E-01	1.03E-01	1.29E-01	1.13E-01
Ru-105+D	3.77E-03	2.10E-02	4.15E-03	3.29E-03	3.07E-03	3.80E-03	3.35E-03
Ru-106+D	8.26E-03	6.14E-01	9.01E-03	7.24E-03	6.77E-03	8.30E-03	7.40E-03
Sb-129+D	4.39E-03	3.13E-02	4.85E-03	3.88E-03	3.63E-03	4.30E-03	3.97E-03
Te-131	9.13E-04	1.61E-02	1.01E-03	7.83E-04	7.26E-04	9.71E-04	7.79E-04
Te-131m+D	6.24E-03	1.73E-02	6.90E-03	5.48E-03	5.12E-03	6.18E-03	5.52E-03
Te-132+D	2.50E-01	7.82E-01	2.76E-01	2.18E-01	2.05E-01	2.49E-01	2.22E-01
Te-133m+D	1.06E-02	5.22E-02	1.17E-02	9.32E-03	8.74E-03	1.04E-02	9.42E-03
Te-134	6.57E-03	9.12E-03	7.22E-03	5.68E-03	5.28E-03	6.81E-03	5.72E-03
I-131 (ELE)	2.47E+00	6.47E+00	2.67E+00	2.12E+00	1.96E+00	2.61E+00	2.18E+00
I-132	2.69E-01	8.91E-01	2.97E-01	2.36E-01	2.22E-01	2.63E-01	2.41E-01
I-133	1.68E+00	1.19E+01	1.83E+00	1.47E+00	1.37E+00	1.69E+00	1.50E+00
I-134	1.24E+00	4.56E+00	1.37E+00	1.09E+00	1.03E+00	1.20E+00	1.11E+00
I-135	2.31E+00	7.90E+00	2.54E+00	2.06E+00	1.93E+00	2.20E+00	2.07E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137+D	2.41E-01	7.64E-01	2.65E-01	2.11E-01	1.97E-01	2.39E-01	2.15E-01
Cs-138	4.42E-02	3.29E-01	4.81E-02	3.93E-02	3.72E-02	4.22E-02	3.97E-02
Ba-139	6.42E-04	1.58E-01	7.10E-04	5.28E-04	4.84E-04	7.62E-04	5.03E-04
Ba-140+D	9.52E-01	4.03E+00	1.04E+00	8.43E-01	7.97E-01	9.22E-01	8.52E-01
Ba-141	1.80E-03	2.02E-02	1.97E-03	1.56E-03	1.46E-03	1.86E-03	1.59E-03
Ba-142	6.33E-04	3.75E-03	7.02E-04	5.55E-04	5.20E-04	6.28E-04	5.59E-04
La-141	9.36E-04	2.49E-01	1.03E-03	8.32E-04	7.88E-04	8.87E-04	8.40E-04
La-142	3.90E-02	1.92E-01	4.17E-02	3.47E-02	3.31E-02	3.67E-02	3.55E-02
Ce-141	3.80E-02	1.19E-01	4.24E-02	3.04E-02	2.76E-02	4.76E-02	2.73E-02
Ce-143	2.06E-02	2.83E-01	2.25E-02	1.72E-02	1.59E-02	2.19E-02	1.70E-02
Ce-144+D	2.59E-02	6.59E+00	2.85E-02	2.18E-02	2.03E-02	2.81E-02	2.06E-02
Pr-143	2.78E-09	6.96E-01	3.06E-09	2.44E-09	2.29E-09	2.73E-09	2.49E-09
Pr-144	6.43E-07	2.83E-04	6.94E-07	5.71E-07	5.41E-07	6.12E-07	5.80E-07
U-234	2.75E-05	2.17E-04	7.87E-06	5.93E-06	3.18E-06	1.01E-05	3.12E-06
U-235+D	2.14E-04	3.28E-04	2.31E-04	1.69E-04	1.54E-04	2.55E-04	1.59E-04
U-238+D	6.44E-07	2.56E-04	6.94E-07	5.11E-07	4.68E-07	7.13E-07	4.47E-07
TOTALS	1.13E+01	6.04E+01	1.24E+01	9.89E+00	9.25E+00	1.12E+01	1.01E+01

Table C-39. (continued).

Report Date: 24-Jun-91
Infant receptor

2/11-24/56 Episodic Release
IET #3

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Ar-41	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	2.03E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	4.03E-36	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	4.20E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	1.81E-01	1.53E-02	1.53E-02	1.53E-02	1.53E-02	1.39E+00	9.90E-01
Sr-91	1.44E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.31E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	6.52E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	2.98E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	4.92E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	3.83E-02	2.52E-03	1.46E-03	2.52E-03	4.21E-03	8.80E-02	2.41E-02
Zr-97	2.30E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-96	2.96E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.05E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.02E-02	1.31E-03	1.46E-03	1.46E-03	2.00E-03	1.86E-03	2.13E-03
Ru-105	1.55E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	5.52E-03	1.37E-03	1.43E-03	1.43E-03	1.43E-03	1.68E-03	2.23E-03
Sb-129	1.74E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131	2.74E-25	0.00E+00	4.52E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	1.44E-05	0.00E+00	1.34E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	2.76E-04	0.00E+00	8.21E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-133m	7.38E-15	0.00E+00	1.55E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-134	9.25E-18	0.00E+00	1.45E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	3.40E+01	1.45E-01	1.14E+03	2.75E-01	1.42E-01	2.04E-01	2.16E-01
I-132	9.11E-08	7.59E-09	2.08E-06	1.52E-08	1.21E-08	1.06E-08	1.21E-08
I-133	1.35E-01	0.00E+00	4.60E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	7.49E-14	0.00E+00	9.07E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	5.63E-04	0.00E+00	1.89E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-129m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	2.69E-02	2.15E-02	2.55E-02	2.42E-02	2.55E-02	2.82E-02	3.63E-02
Cs-138	1.63E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	5.63E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.52E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-141	1.11E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-142	3.19E-49	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-141	7.23E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	4.11E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.08E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-143	2.90E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	4.08E-02	8.15E-04	8.15E-04	8.15E-04	1.22E-02	3.51E-02	4.49E-02
Pr-143	1.42E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	2.80E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-234	1.43E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.26E-02	1.49E-03
U-235	4.37E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.47E-04	4.37E-05
U-238	3.73E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.00E-06	4.06E-07
TOTALS	3.45E+01	1.88E-01	1.15E+03	3.20E-01	2.03E-01	1.77E+00	1.32E+00

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APPENDIX D
DISCUSSION OF UNCERTAINTIES

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D.1 UNCERTAINTY AND CONSERVATISM IN HISTORICAL DOSE CALCULATIONS

D.1.1 Importance of Uncertainty Estimates

All offsite dose estimates have associated uncertainties. This is due, in part, to the fact that these estimates are necessarily the result of a series of modeling calculations rather than direct measurements. Even for a preliminary evaluation of offsite dose it is important to recognize that these uncertainties exist and to have some appreciation of their magnitude and how they arise. Knowledge of the uncertainties is valuable for understanding how much reliance to place in the dose estimates and for anticipating areas of potential public concern.

If a more detailed assessment of historical doses is to be done following this preliminary evaluation, then some treatment of uncertainty will be an important element of the assessment. Every major historical dose reconstruction project has had to address this problem. Two such efforts have been recently completed, one for the atmospheric weapons testing program at the Nevada Test Site and one for the Japanese atomic bomb survivors. Both found that a rigorous uncertainty analysis was required for reasons of technical credibility and public relations. Another major effort is currently under way at Hanford Site to estimate doses to the public over the historical lifetime of that Site. The Hanford project has identified uncertainty as an important concern and is making every effort to reduce uncertainties to a minimum and to quantify their magnitudes.

D.1.2 Nature of Uncertainties

There are two primary types of uncertainty in calculations of this type. The first results from a lack of precise knowledge of important parameter values such as amount of radioactivity released and exact meteorological conditions. When such information is missing, assumptions must be made that inevitably introduce bias in the calculations. These uncertainties can, in principle, be reduced by obtaining more precise information. The preferred

method for improving the data would be to take additional measurements, but this is impossible to do for events that have occurred in the past. Therefore, the current evaluation must depend entirely on available historical data. This information is necessarily incomplete, if only because it was obtained for reasons other than estimating offsite doses.

The second type of uncertainty results from the inherent variability of the processes that transport radioactivity offsite and result in potential doses. Atmospheric dispersion, deposition onto the ground, and movement through the food chain are examples of processes with inherent uncertainties. These uncertainties are very difficult to reduce, but they can be quantified by any one of several approaches to formal uncertainty analysis. A formal uncertainty analysis can be labor intensive and time consuming; therefore, the resources devoted to uncertainty analysis should be balanced with those applied to the rest of the calculation.

No formal uncertainty analysis was done for this preliminary evaluation, however, some useful generalizations can be made concerning uncertainty in historical dose estimates. The following sections briefly describe the types and levels of uncertainty to be expected in each of the major portions of an offsite dose calculation.

D.1.3 Source Term

Source term information includes the amounts of radionuclides released; the physical and chemical form of the radionuclides released; the physical description of the release (e.g., stack height and exit velocity); and the precise timing of releases. Uncertainties in release estimates tend to introduce a systematic bias in dose estimates, that is a consistent overestimate or underestimate, rather than a random error. Uncertainties in the source term often dominate the total uncertainty in evaluations of this kind. This is partly because they can often be quite large but also because all subsequent steps of the dose calculation use the release estimate as input.

It is difficult to estimate the magnitude of these uncertainties for historical releases at the Idaho National Engineering Laboratory (INEL). In most cases, adequate information is not available. However, there are some general patterns that can be expected. Source term estimates for recent years are more precisely known than for earlier years. This is largely because the technology for monitoring releases has continually improved over the last several decades so that direct measurements of source terms have become more detailed and precise. For earlier years, where direct measurements are not available, releases must be calculated on the basis of process and operational information in various project reports. Documentation of experiments, tests, and reactor programs is generally more complete and more readily available for recent years than for earlier years.

For any given operational or episodic release, estimates of total activity released are better known than are estimates for individual radionuclides. There are two reasons for this. First, some estimates of releases are necessarily based only on measurements of total radioactivity. This is particularly true of measurements made during the 1950s and early 1960s when monitoring equipment was incapable of distinguishing among individual radionuclides. In this case, where the total activity is known but the mix of radionuclides is not, additional calculations and supporting assumptions are required to estimate the releases of individual radionuclides. These calculations and assumptions introduce additional uncertainty. Second, even with the best monitoring equipment, some radionuclides are much more difficult to detect and measure than others. This results in relatively greater uncertainty in estimates of releases for the radionuclides that are more difficult to measure.

One consequence of this variation in source term uncertainty among radionuclides is that uncertainties associated with individual organ doses vary somewhat whenever these doses are strongly influenced by single radionuclides. For example, large uncertainties in release estimates for I-131 would increase the uncertainty in thyroid doses but would have little influence on the uncertainty in dose estimates to the bone surface. Uncertainties in Sr-90 releases would have the opposite effect, contributing

to the uncertainty in bone surface doses while having no effect on uncertainty in thyroid dose.

With regard to total dose, it is often the case that the smallest relative uncertainties are associated with those radionuclides that make the largest contributions to total dose. This is partly because the properties of radionuclides, such as energetic gamma-ray emission and relatively long half-life, that cause them to deliver high doses also make them comparatively easy to measure in facility effluents. Additionally, certain radionuclides (e.g., I-131, Cs-137, and Sr-90) that are known to be major dose contributors were routinely monitored and reported even during the 1950s and 1960s. Therefore, the data used to develop release estimates for these radionuclides are often of good quality and few additional calculations are needed.

In contrast, radionuclides that are difficult to measure in effluents, either because they have very short half-lives or emit very low-energy radiations, have large uncertainties in release estimates. They also are often only small contributors to total dose. Tritium is an example of a radionuclide that is very difficult to measure because of its low energy. Therefore, the uncertainties associated with tritium releases and resulting doses are comparatively large, but the doses themselves are small.

D.1.4 Meteorology and Atmospheric Dispersion

Atmospheric transport is the dominant pathway for offsite doses at the INEL. Good meteorological data are available from the National Oceanic and Atmospheric Administration for recent years, and reasonable extrapolations can be made for the early decades of INEL Site operation. Even with the best available data, however, there are inherent uncertainties in atmospheric dispersion calculations. For the long-term average conditions of the type that can be applied to annual average operational releases, uncertainties of a factor of 2 or 3 are common. For the short-term conditions applicable to accidental or episodic releases, uncertainties are always larger, often by as much as another factor of 10. With short-term releases, the concentrations within the plume or puff are variable and are not "averaged" by the addition

of many puffs at a single location. Also more uncertain is whether the plume passes over or just misses a receptor location.

D.1.5 Dose Calculation

The uncertainty in estimated dose varies with the exposure mode. External exposure from immersion in contaminated air has the smallest uncertainty and internal dose from ingestion has the largest uncertainty. All doses are calculated from air and ground concentrations estimated from the atmospheric dispersion calculations and have uncertainties at least as large as those associated with estimated air concentrations. External exposure from cloud immersion and inhalation dose are the most straightforward and least uncertain calculations because they depend directly on air concentration only. Air concentration is given by the product of the source term estimate and the atmospheric dispersion factor and reflects the uncertainty of both those steps in the calculation. External exposure from contaminated ground surfaces and ingestion dose require estimating local deposition on the ground. This process alone is often not known to within a factor of 2. Ingestion doses are also complicated by an involved calculation of radionuclide movement through the food chain that is subject to large uncertainties that have ranged between factors of 3 and 10 for other dose assessments.

D.1.6 Total Uncertainty

The net effect of these uncertainties in source term, atmospheric dispersion, and dose calculation varies with radionuclide, with organ, with age group, and from year to year over the history of the INEL. It is difficult to estimate precise values with the available information, but it would not be surprising to find an overall uncertainty in the range of a factor of 5 in total dose and a factor of 10 for some of the earlier years.

D.2 CONSERVATISM IN HISTORICAL DOSE ESTIMATES

In the context of estimating potential radiation dose, a conservative calculation is one that includes assumptions or parameter values that tend to bias the result toward an overestimate. The principal reason for introducing conservatism into dose calculations is to ensure that they are less likely to under estimate potential doses than to over estimate them. Conservative calculations are often used to take into account calculational uncertainties where no formal uncertainty analysis has been done. The intent is to estimate a dose that is large enough to encompass any uncertainties because of systematic bias introduced by assumptions that must be made because data are lacking.

The most extreme example of a conservative estimate is the worst case analysis in which all parameters and assumptions are chosen to maximize the estimated dose. Worst case analyses have the advantage that they place an absolute upper bound on potential dose but they are otherwise of very limited utility. A worst case analysis usually obscures the relative importance of different steps in the calculation and does not allow useful comparison of doses estimates for different sets of circumstances. These limitations make a worst case approach inappropriate for use in a historical dose evaluation where comparisons from year to year and place to place are important.

The approach used in this evaluation was to make the most realistic estimate of source term releases and atmospheric dispersion that the data allow. This means that estimates of integrated air concentration at each receptor location are realistic and may not always be conservative. However, conservative assumptions were deliberately incorporated into the parameters and assumptions used in the dose calculation scenarios. This isolates much of the conservative bias to the final step in the calculation and allows the degree of conservative bias to be explicit. If a future study chooses to use a different set of assumptions regarding exposure scenarios it should still be able to make use of much of the other information in this report.

Conservative assumptions built into the dose calculations affect each route of exposure differently. They show a pattern similar to the degree of

uncertainty associated with each dose mode. External dose from contaminated ground surfaces, for example, assumes continuous residence at the receptor location for 50 years following deposition. External dose from immersion in air assumes continuous presence and no shielding from terrain, clothing, or structures. Internal dose from inhalation of contaminated air assumes continuous presence at the receptor location during the period of exposure and a conservatively high average breathing rate.

The calculation of ingestion dose is more complex than the other three dose modes and contains a series of conservative assumptions and parameter values. For example, episodic releases were evaluated assuming that all milk came from a dairy cow grazing at the receptor point even at times and locations where such a cow would be unlikely or impossible. In all cases, it was assumed that the sole source of fresh vegetables was a garden located at the receptor point even though this is unlikely or impossible for many times and locations. In addition, the dietary intakes assumed for all foods and all age groups were the worst case conservative diets suggested by the U. S. Nuclear Regulatory Commission. These assumptions were made to avoid missing a pathway that might have made a significant contribution to offsite dose. The result is that estimates for ingestion dose are likely to be more conservative than those for the other dose modes.

Not all conservative assumptions in this study are confined to the dosimetric stage of the calculation. In some cases realistic estimates of source term releases or atmospheric dispersion factors could not be reliably derived from the available data. This is particularly true for the early years of INEL Site operation. When data were not available, intentionally conservative assumptions were made to avoid underestimating potential doses. The Initial Engine Test (IET) #3 release is the most dramatic example of this. Available operational data did not support a complete analysis of either the reactor inventory or the release fractions for groups of radionuclides; therefore, conservative assumptions were made in each case. Particulate emissions during IET #3 were estimated to be 4000 Ci based on measured effluent releases. However, this estimate was judged to be quite uncertain. Conservative assumptions were made, and the particulate release used in the dose calculations was 42,500 Ci, which was over 10 times greater than the

releases estimated from effluent measurements. Meteorological data at IET were also lacking for the period of the IET #3 release. The atmospheric dispersion coefficient was conservatively calculated assuming that the wind blew directly toward the nearest receptor to the northeast when releases of radioactivity occurred. As a consequence, doses estimated for IET #3 are the highest reported in this evaluation. This high degree of conservatism reflects the lack of pertinent monitoring data and the resulting uncertainty in each step of the calculation. If more complete information were to be discovered in test reports or Government archives, more realistic estimates could be made and the estimated doses from IET #3 would be likely to decline.

In this preliminary evaluation every effort has been made to choose assumptions and parameter values so that these estimates provide a realistic but conservative estimate of potential doses. It is anticipated that any further refinements of these estimates will result in lower dose values. With respect to uncertainty this means that, while some doses may be uncertain within a factor of 10, they are likely to be 10 times smaller and unlikely to be very much larger.

APPENDIX E

ENVIRONMENTAL SURVEILLANCE

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E.1 INTRODUCTION

This appendix presents an overview of the environmental surveillance programs conducted at the Idaho National Engineering Laboratory (INEL) over the 40-year history of the INEL Site. The referenced reports should be consulted for further details. This appendix covers ambient environmental surveillance. Effluent monitoring results are included in Appendix A. This appendix focuses on the surveillance performed by the Radiological and Environmental Sciences Laboratory (RESL), formerly the Health Services Laboratory (HSL), and the Idaho Health and Safety Division of the U. S. Atomic Energy Commission (AEC). Environmental surveillance is also performed inside facility fences by contractor organizations. Recent characterization studies have been conducted under the Consent Order and Compliance Agreement with the U. S. Environmental Protection Agency (EPA) for hazardous waste, but that surveillance is not directly related to radiation doses to people living near the INEL Site. This appendix does not cover nonradiological environmental surveillance.

Some of the many environmental reports published throughout the history of the INEL Site are listed in Section E.2. The present radiological environmental surveillance program is discussed in Section E.3. A brief history of the program is presented in Section E.4. Selected offsite environmental surveillance results are compared to INEL Historical Dose Evaluation results in Section E.5.

E.2 ENVIRONMENTAL REPORTS

Many environmental reports have been published by the AEC, the U. S. Department of Energy (DOE), and its contractors. All of the reports referenced in this section have been available to the general public. Many of them were distributed to Federal, state, and local leaders. Some of the earliest reports were published by the Health and Safety Division. The Health and Safety Division of the AEC published a series of reports starting with IDO-12000 (March 1952) and IDO-12001 (November 1951). Six annual reports for the Division were published (AEC, 1959; AEC, 1960a; AEC, 1961a; AEC, 1962a; AEC, 1963a, AEC, 1964a). The HSL was organized in May 1967 from six branches of the Health and Safety Division and continued the same report series. Four laboratory annual reports were published (AEC, 1969a; AEC, 1970a; AEC, 1971a; AEC, 1972a). The name of the HSL was changed to RESL in 1978. The report series was changed to DOE/ID-12098 in June 1983. The report series included some of the radioecology research reports of the HSL, RESL, and some National Oceanic and Atmospheric Administration (NOAA) reports, although many of those studies have been published in peer-reviewed journals. Environmental surveillance programs at the INEL Site have been described in annual environmental reports since 1959, although a formal report number was not assigned until 1976 [IDO-12082(76)]. The reports are listed individually in the Section E.6 of this appendix (AEC, 1960b; AEC, 1961b; AEC, 1962b; AEC, 1962c; AEC, 1963b, AEC, 1964b; AEC, 1964c; AEC, 1965a; AEC, 1965b; AEC, 1966a; AEC, 1966b; AEC, 1967a; AEC, 1967b; AEC, 1968a, AEC, 1968b; AEC, 1969b; AEC, 1969c; AEC, 1970b, AEC, 1970c; AEC, 1971b; AEC, 1972b; AEC, 1973; AEC, 1974; Chew and Mitchell, 1988; DOE, 1978a; DOE, 1979; DOE, 1980; DOE, 1981; DOE, 1982; DOE, 1983a; ERDA, 1975; ERDA, 1976a; ERDA, 1977a; Hoff et al., 1984; Hoff et al., 1985; Hoff et al., 1986; Hoff et al., 1987; Hoff et al., 1989; Hoff et al., 1990). For the years 1959 through 1973, the data were published in the U. S. Department of Health, Education, and Welfare report, *Radiological Health Data* (HEW, 1960; HEW, 1961a; HEW, 1961b; HEW, 1962a; HEW, 1962b; HEW, 1963; HEW, 1964; HEW, 1965a; HEW, 1965b; HEW, 1966a; HEW, 1966b; HEW, 1967a; HEW, 1967b; HEW, 1968a; HEW, 1968b; HEW, 1969; HEW, 1970; EPA, 1971a; EPA, 1971b; EPA, 1973; 1974a; EPA, 1974b; EPA, 1974c). Before this period, environmental data for individual programs and tests were published in AEC and contractor reports. U. S. Geological Survey (USGS) reports have been

published since 1951 in a report series numbered sequentially starting with IDO-22000. In 1986, the report series prefix changed to DOE/ID while the numbering sequence continued (i.e., DOE/ID-22066). Summaries of environmental monitoring at the INEL Site have also appeared in other publications (EPA, 1974b; ERDA, 1977b; Bowman, 1984; Smith et al., 1986). INEL contractors also issue reports of environmental surveillance at their facilities. For instance, EG&G Idaho, Inc. has published an annual report for their waste management areas at the INEL Site since 1974.

E.3 PRESENT ENVIRONMENTAL SURVEILLANCE PROGRAM

Environmental surveillance programs at the INEL Site are conducted to determine whether the public and the environment are adequately protected during DOE operations and whether operations are in compliance with DOE and other applicable Federal, state, and local radiation standards and requirements. The programs are also conducted to detect and quantify unplanned releases. Environmental impacts are determined by measuring radioactivity in the environment, where such measurements are possible, or by modeling the transport of radioactive substances through environmental pathways in cases where environmental concentrations are too low to measure. Measurements on the INEL Site or at the INEL Site boundary are compared to similar measurements at background or control locations in an attempt to determine whether measured radioactivity is from INEL Site operations. All measured concentrations are compared to applicable environmental standards. Where radioactive concentrations are high enough to be measured regularly, long-term trends are presented.

The RESL and USGS sample pathways by which radioactive effluents from INEL Site operations could reach nearby populations. The sampled pathways are air, water, foodstuffs, soil, direct radiation, and game animals. Thus, the sampled pathways include those that might potentially lead to inhalation of contaminated air, ingestion of contaminated food or water, and direct exposure to penetrating radiation. There is no significant movement of surface water from onsite to offsite locations. Ground water is monitored, but radioactive waste released to the aquifer has never been observed in drinking water wells beyond the INEL Site boundary.

Important INEL effluents are generally monitored at two points in each environmental pathway--usually at the point of release and at a point in the exposure pathway as near the receptor (man) as possible.

Most of the environmental surveillance programs involve collecting representative samples and performing laboratory analyses. The analytical methods for environmental samples are carefully reviewed to verify that analyses are made with sufficient sensitivity to verify compliance with

appropriate standards. High reliability is obtained by a stringent quality assurance program. Quality assurance is outlined in the annual INEL Site Environmental Report (Hoff et al., 1990). The RESL is directly traceable to the National Institute of Standards and Technology (NIST) for radiological analyses.

Independent verification of the INEL's environmental surveillance is currently conducted by the EPA, the State of Idaho, and Idaho State University. Samples of air, precipitation, drinking water, and milk from Idaho Falls and water from the Snake River near Buhl, Idaho, are analyzed by the EPA's National Air and Radiation Environmental Laboratory (EPA, 1990). Since 1977, under a working agreement between the State of Idaho and DOE, environmental samples collected by RESL or USGS may be divided between the State and the RESL or USGS. Water samples have been split under that agreement. In 1990, DOE and the State of Idaho signed an Environmental Oversight and Monitoring Agreement that provides for oversight and monitoring of the INEL by the State. Since July 1989, Idaho State University has conducted independent environmental monitoring of air, water, and soil on and around the INEL Site.

A discussion of each major routine surveillance program currently implemented follows. The programs are summarized in Table E-1 and the frequency of analysis for all media and the number of locations sampled is shown. All of the air samplers and thermoluminescent dosimeters (TLDs) used in the routine surveillance program are collecting data continuously. Gross counting of samples is used for establishing trends or for screening groups of samples. However, the most sensitive indicators of radiological impacts of INEL Site operations are the analyses of samples for individual radioisotopes. The current minimum detectable concentrations shown in Table E-1 for most radioisotopes permit calculation of dose commitments to the public of 0.1 mrem/yr or less. This represents about 1% of the current EPA airborne emission standard.

Table E-1. RESL radiological environmental surveillance program summary (1989)

Medium Sampled	Type of Analysis	Number of Locations and Frequency: Onsite	Number of Locations and Frequency: Offsite	~Minimum Detectable Concentration
Air	**Low-Volume Samplers:**			
	Gross alpha	3 weekly	3 weekly	3.0E-16 µCi/mL
	Gross beta	12 weekly	11 weekly	8.0E-15 µCi/mL
	Specific gamma	12 quarterly	4 quarterly	1 to 10.0E-15 µCi/mL
	Pu	6 quarterly	4 quarterly	6.0E-18 µCi/mL
	Am	6 quarterly	4 quarterly	8.0E-18 µCi/mL
	Sr-90	2 quarterly	4 quarterly	1.0E-16 µCi/mL
	High-Volume Samplers:			
	Gross gamma	2 daily	--	N/A[a]
	Specific gamma	2 monthly	--	1 to 10.0E-16 µCi/mL
	Kr-85 Sampler	1 biweekly	--	2.0E-12 µCi/mL
	Tritium Samplers:			
	H-3 as HTO	2 at 1 to 2/quarter	1 to 2/quarter	1.0E-11 µCi/mL
Water	**Drinking Water:**			
	Gross alpha	26 monthly	13 semiannually[b]	3.0E-9 µCi/mL
	Gross beta	26 monthly	13 semiannually	4.0E-9 µCi/mL
	H-3 as HTO	26 monthly	13 semiannually	4.0E-7 µCi/mL
	Sr-90	2 monthly	--	5.0E-10 µCi/mL
Animal Tissues[c]	**Beef--Muscle, Liver:**			
	Specific gamma	2 biennially	2 biennially	7.0E-9 µCi/g
	Pu-239/240	2 biennially	2 biennially	5.0E-11 µCi/g
	Am-241, Pu-238	2 biennially	2 biennially	1.0E-10 µCi/g
	Sheep--Muscle, Liver:			
	Specific gamma	4 annually	2 annually	7.0E-9 µCi/g
	Game Animals--Muscle:			
	Specific gamma	Varies annually[d]	--	7.0E-9 µCi/g
Foodstuffs	**Milk:**			
	I-131	None produced	1 weekly	1.0E-9 µCi/mL
	I-131	None produced	10 monthly	1.0E-9 µCi/mL
	Sr-90	None produced	10 annually	2.0E-9 µCi/mL
	H-3 as HTO	None produced	10 annually	4.0E-7 µCi/mL
	I-129	None produced	3 annually	3.0E-10 µCi/mL
	Wheat:			
	Specific gamma	None produced	10 annually	4.0E-9 µCi/mL
	Sr-90	None produced	10 annually	4.0E-9 µCi/mL
	Lettuce:			
	Specific gamma	None produced	8 annually	2.0E-7 µCi/mL
	Sr-90	None produced	8 annually	8.0E-8 µCi/mL
Soil	Specific gamma	Varies annually[e]	12 biennially	4.0E-8 µCi/mL
	Pu	Varies annually	12 biennially	2.0E-9 µCi/mL
	Am	Varies annually	12 biennially	3.0E-9 µCi/mL
	Sr-90	Varies annually	12 biennially	9.0E-8 µCi/mL

Table E-1. (continued)

Medium Sampled	Type of Analysis	Number of Locations and Frequency: Onsite	Number of Locations and Frequency: Offsite	Minimum Detectable Concentration
Direct Radiation Exposure	Thermoluminescent Dosimeters	135 semiannually	13 semiannually	5 mR
	Gamma Radiation Surveys	Varies annually[f]	--	N/A

a. Not applicable.

b. Two additional offsite drinking water samples and three surface water samples from springs in the Magic Valley area are split monthly with Idaho State University Environmental Monitoring Group.

c. "Onsite" animals grazed onsite for a least 4 weeks before being sampled. "Offsite" animals have never grazed onsite and serve as controls.

d. Only road-killed game animals are sampled onsite. No controls are generally collected except for specific ecological studies.

e. Onsite soil sampling is performed each year at different onsite facilities on a rotating 7-year schedule.

f. Surveys are performed each year at different onsite facilities on a rotating 3-year schedule.

E.3.1 Air Sampling

Air concentrations are monitored by continuously pumping ambient air through a filter that traps the radioactive particulates. The filters are removed for laboratory measurement at set intervals. Low-volume air filters are changed weekly (see Figure E-1 for locations). A prefilter collects particulates, while a charcoal paper filter samples for iodine gases. The particulate filters are also composited quarterly for increased sensitivity. Low-volume (40 L/min) air samplers are used to quantify airborne radionuclide concentrations at locations onsite and offsite, primarily where groups of people are located. Statistical comparisons are made between individual onsite locations and distant locations, between onsite locations as a group and distant locations, and between boundary and distant locations.

High-volume (1000 L/min) air samplers at the Experimental Field Station (EFS) and at the Central Facilities Area (CFA) provide rapid indications of airborne radioactivity not detected by facility air monitors. These filters, changed each working day, are composited monthly for specific radionuclide

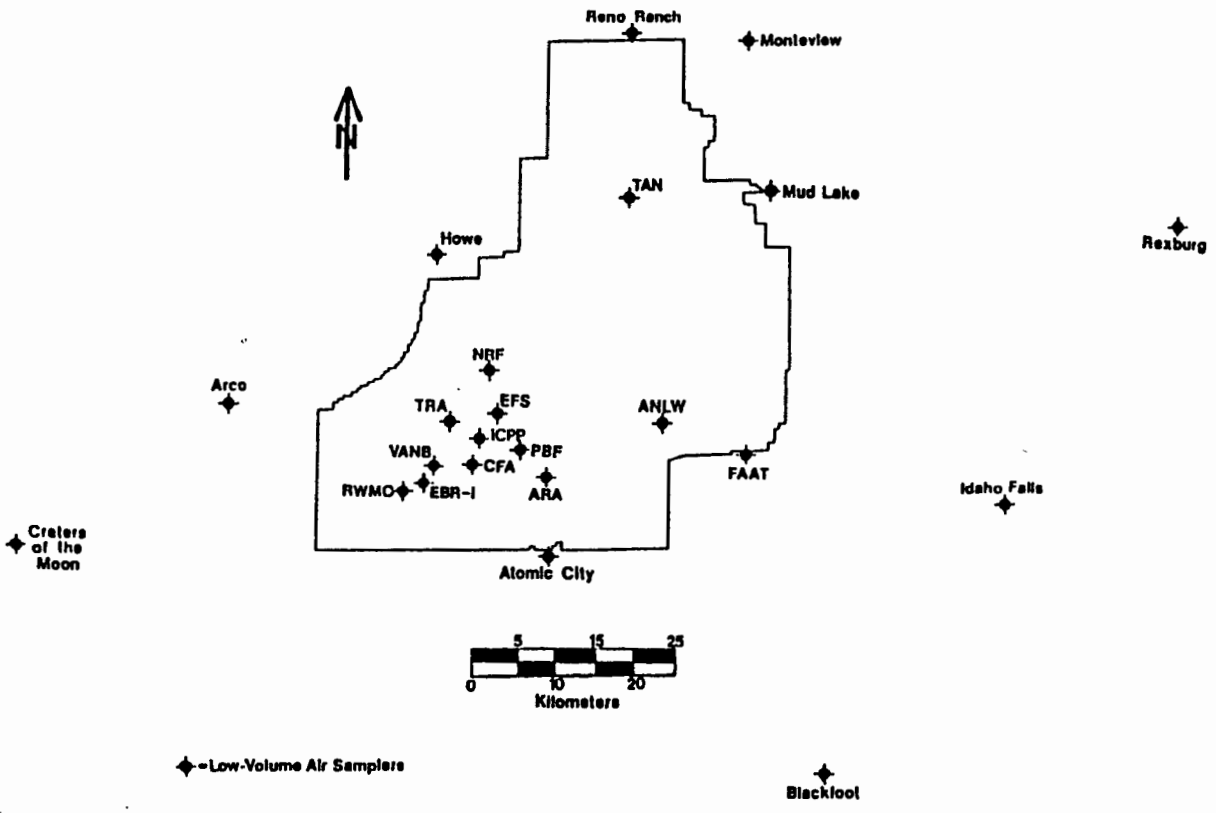


Figure E-1. INEL Site and vicinity air sampling network.

analysis. On the initial gross gamma analysis, concentrations of naturally occurring radon and thoron daughters are often quite high and would likely mask any manmade radioactivity that could potentially be present from INEL Site operations. Several gross gamma analyses are subsequently performed and decay curves are plotted over the first day to determine if manmade radioactivity is present in the sample. If there is any indication of radioactivity other than from natural sources or fallout, the filter is analyzed by gamma spectrometry for specific radionuclides.

Tritium samplers are located at EFS, Van Buren (VANB), and Idaho Falls (see Figure E-1). Air moisture is trapped by a column filled with silica gel. Water is driven off of the silica gel, collected, and analyzed for tritium.

A sampler at CFA collects a continuous sample of air in a large Tedlar bag. The air is compressed into a cylinder every 2 weeks. Two cylinders are shipped monthly to the EPA's Environmental Monitoring Systems Laboratory in Las Vegas, Nevada, for analysis of Kr-85 concentrations. The program began in January 1984.

E.3.2 Water Monitoring

The USGS performs most of the ground water and surface water monitoring on and near the INEL Site. Monitoring of drinking water and some surface water sampling is performed by RESL and facility contractors. The USGS maintains more than 100 aquifer observation wells on or near the INEL Site. More than 100 auger holes are also available for sampling shallow perched water. Water levels in the wells and various radiological and nonradiological components of the aquifer are monitored. The purpose of monitoring is to determine whether or not changes to the general hydrologic conditions of the region have been caused by INEL operations.

The USGS radiological ground water monitoring program is summarized in Table E-2. The type, frequency, and depth of sampling depends upon the information needed in a specific area. Specific gamma analyses are performed for Cr-51, Co-60, Cs-137, and any other gamma emitters that are detected in the water. The locations and frequency of the sampling are shown in

Table E-2. USGS radiological ground water monitoring program summary (1989)

Type of Analysis	Frequency of Analysis	Number of Samples	~Minimum Detectable Concentration (μCi/mL)
Gross alpha	Semiannually[a]	5	3.0E-9
Gross beta	Semiannually[a]	5	4.0E-9
Tritium[b]	Quarterly	48	4.0E-7
Tritium	Semiannually[a]	82	4.0E-7
Specific gamma	Quarterly	9	1 to 10.0E-8[c]
Specific gamma	Semiannually	17	1 to 10.0E-8
Specific gamma	Annually	26	1 to 10.0E-8
Sr-90	Quarterly	27	5.0E-9
Sr-90	Semiannually	26	5.0E-9
Am	Quarterly	4	5.0E-11
Am	Semiannually	4	5.0E-11
Pu	Quarterly	4	4.0E-11
Pu	Semiannually	4	4.0E-11
I-129	~5 years	20-35	6.0E-11

a. In addition, one offsite well is sampled annually.

b. Tritiated water.

c. Varies depending upon radionuclides present in the sample.

Figures E-2 and E-3. Surface water samples are collected quarterly in Idaho from the Big Lost River near Moore, Birch Creek near Blue Dome, the Little Lost River near Howe, and Mud Lake near Terreton.

Offsite drinking water samples are collected semiannually by RESL (see Figure E-4). Onsite drinking water samples are collected monthly from any well in use (currently about 26 wells). RESL analyzes these samples for gross alpha, gross beta, and tritium. The Idaho Chemical Processing Plant (ICPP) samples are also analyzed for Sr-90. INEL contractors analyze the drinking water supplies for nonradiological contaminants.

RESL samples the Snake River at Idaho Falls and Bliss, Idaho, semiannually (see Figure E-4). The sampling locations are upstream and downstream from where water from the Snake River Plain aquifer enters the Snake River. Three of those points of entry are now sampled quarterly by RESL and Idaho State University. The samples are analyzed for gross alpha, gross beta, and tritium.

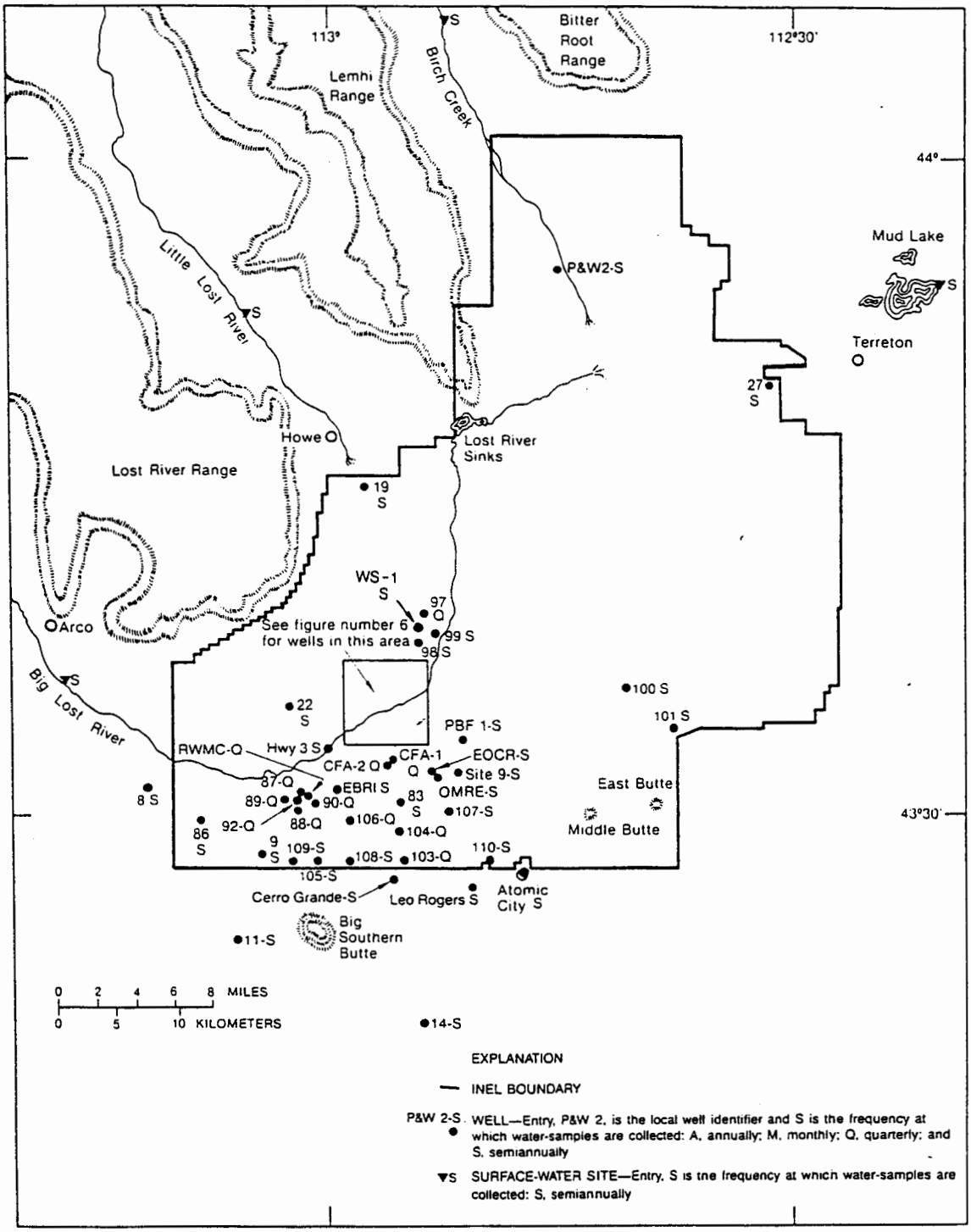


Figure E-2. Locations of wells and frequencies of water-sample collections at the INEL.

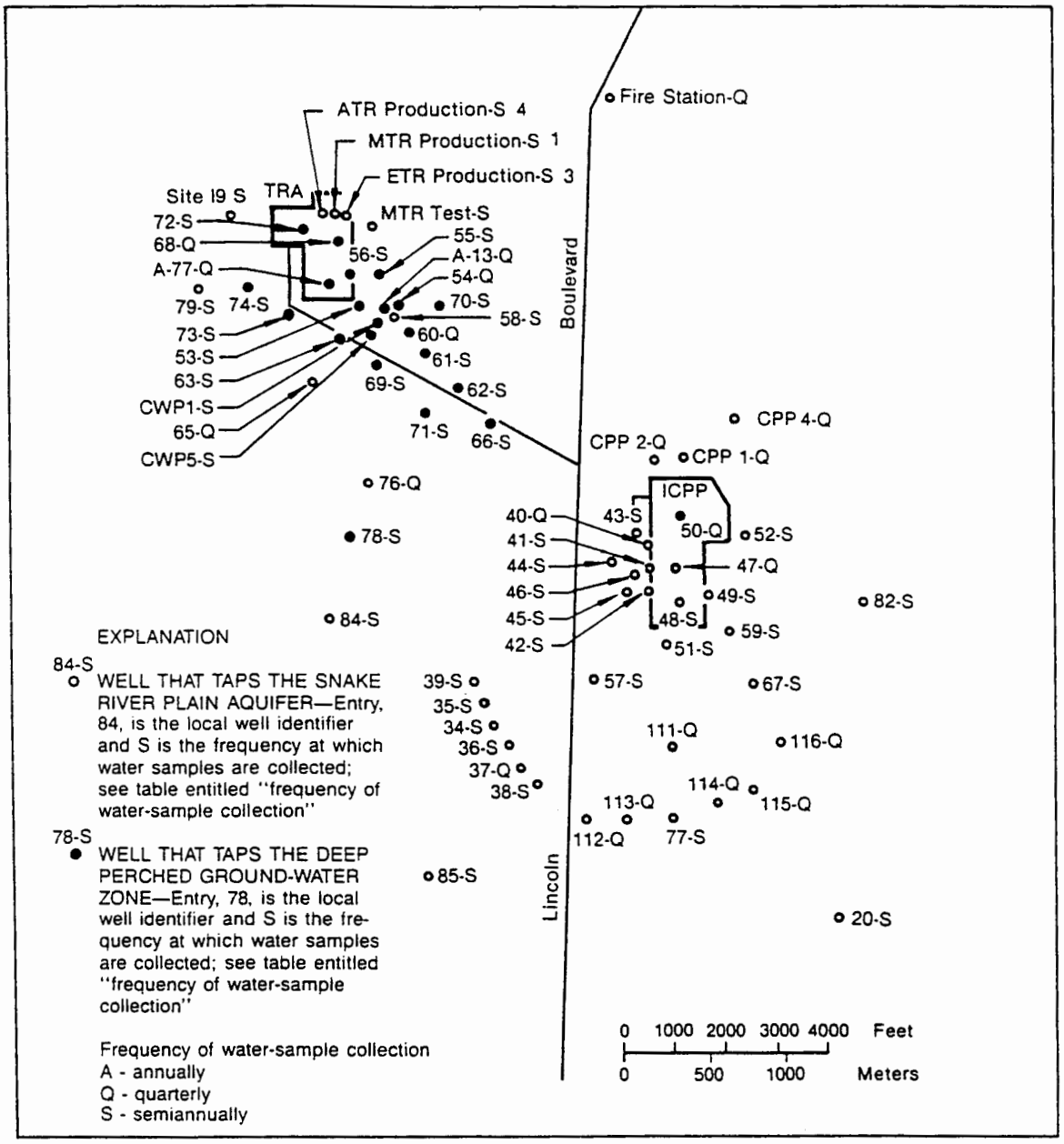


Figure E-3. Locations of wells and frequencies of water-sample collections in the TRA-ICPP area.

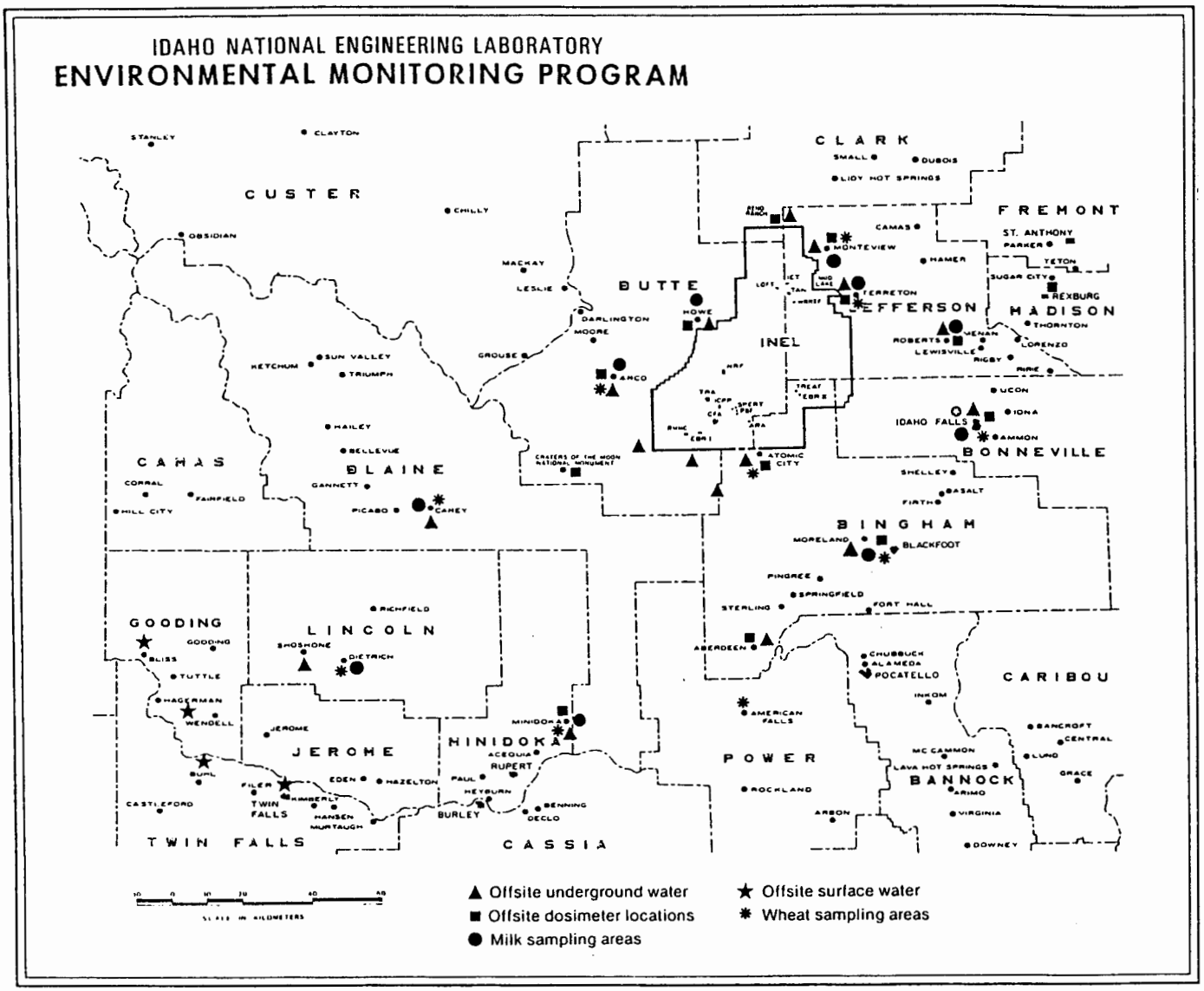


Figure E-4. Boundary and distant water, milk, and wheat sampling locations and environmental dosimeter locations.

E.3.3 Animal Tissues

The land around the INEL Site is used primarily for grazing. In addition, about 60% of the INEL Site is open to grazing. Sheep that have grazed onsite (as well as control sheep) are sampled each year, and cattle are sampled every other year.

Every year, muscle and liver samples are taken from sheep that graze onsite in the southern and northeastern grazing areas as the sheep leave the INEL Site. Every other year, muscle and liver samples are taken from beef cattle that have grazed near the Radioactive Waste Management Complex (RWMC). Concentrations of radionuclides are compared to those measured in control samples.

Hunting and fishing are not allowed on the INEL Site. Game animals do migrate on and off the INEL, representing a potential (but small) exposure pathway. Game animals accidentally killed on INEL roads are sampled. However, the major work on quantifying radionuclides in game animals and potential doses to people harvesting animals has been performed as a part of more intensive research projects.

E.3.4 Foodstuff Monitoring

About 35% of the land adjoining the INEL Site is used for agricultural crops. No crops are grown on the INEL Site. Consequently, foodstuffs are collected near to and distant from the INEL Site, and the radionuclide concentrations are compared. Milk, wheat, and lettuce are sampled routinely. Milk samples are obtained weekly from Idaho Falls and monthly from a broad area around the INEL Site (see Figure E-4). Most milk samples represent bulk milk routes that each cover a relatively small area and are a long distance from the INEL Site. If higher-than-normal levels are found in bulk samples, additional samples are obtained from individual farms. Much of the milk produced in the area (Grade B) is used for cheese; therefore, I-131, which has a short half-life, is less important in this milk product. All milk samples are specifically analyzed for I-131 in theory. The delay between collection

and analysis is probably too long. Each location is analyzed for Sr-90 and tritium once each year. Three milk samples each year are usually analyzed for I-129.

Wheat is collected annually from distant and boundary locations (see Figure E-4). Samples are analyzed by gamma spectrometry and for Sr-90. Past results have shown that 90% of the Sr-90 activity is in the hull, and only 10% is in the edible portion of the wheat.

Lettuce from Arco, Atomic City, Blackfoot, Carey, Howe, Idaho Falls, Mud Lake, and Pocatello, Idaho, is collected when available and analyzed for gamma-emitting radionuclides and Sr-90.

E.3.5 Soil Sampling

Soil samples have been collected from undisturbed distant and boundary locations between 1970 and the present to establish background levels of natural and fallout radioactivity in surface soil and to assess any potential buildup of radioactivity from INEL Site operations. Figure E-5 shows the locations of the soil samples collected biennially since 1978. Soils from each major facility are sampled once every 7 years. The facility soil sampling includes enough locations to establish the extent and distribution of contamination outside facility fences. All soil samples are analyzed for gamma-emitting radionuclides. Most are also analyzed for Sr-90 and alpha-emitting radionuclides.

E.3.6 Direct Radiation Exposure

TLDs are maintained at distant, boundary, and onsite locations listed in Table E-3. See Figure E-4 for distant and boundary locations. Measurements at the distant locations are considered to be from background radioactivity. Measurements on the INEL Site are made along major highways and at each major facility or area. The TLDs consist of five lithium fluoride dosimeter ribbons in a laminated package usually placed about 1 m (3 ft) above ground level. The chips are exchanged every 6 months. An increase of about 10% of the DOE

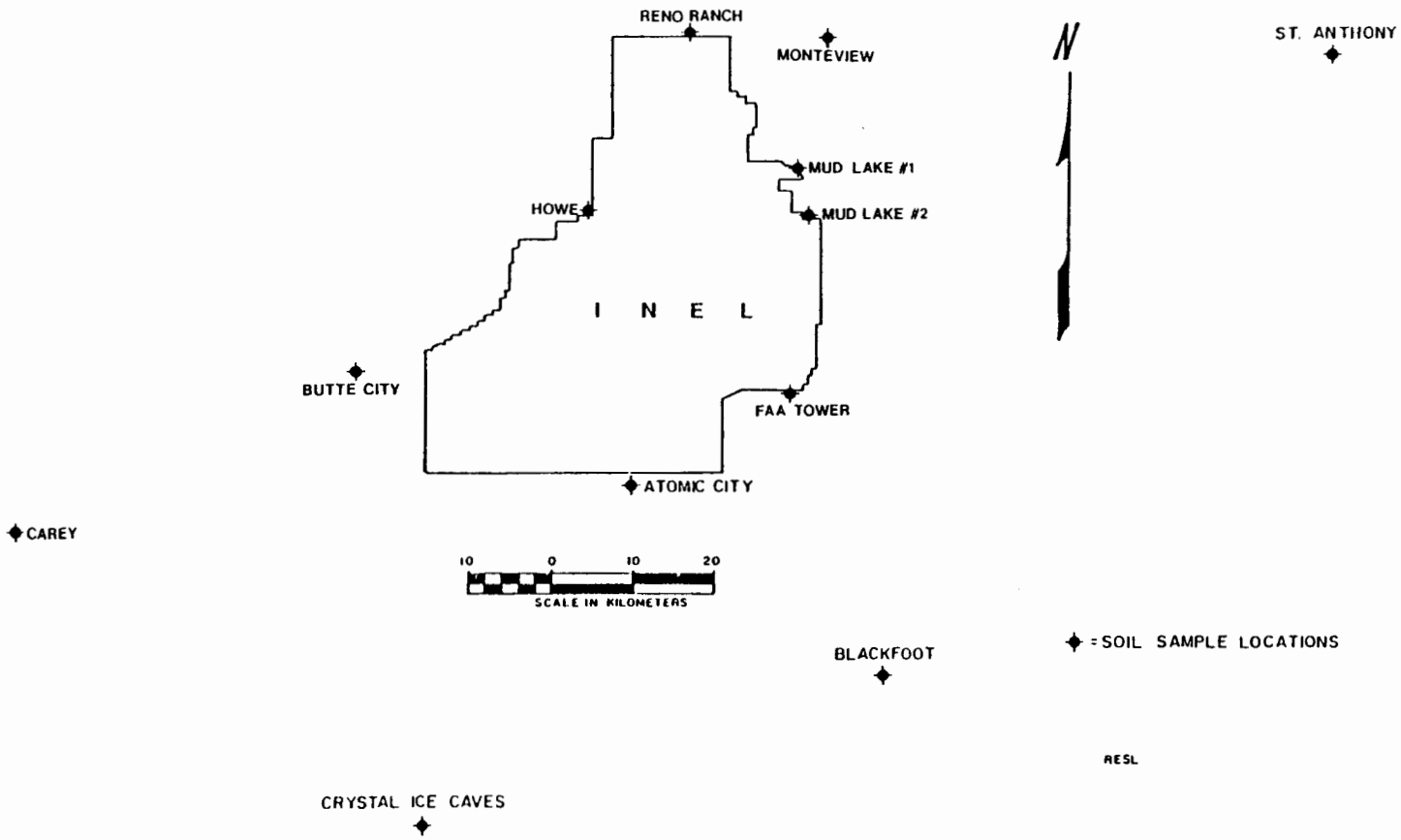


Figure E-5. Soil sampling locations for the INEL vicinity.

Table E-3. Summary of RESL onsite environmental TLD locations

Area	Number of Locations
ARA-I and ARA-II	4
SPERT-PBF	6
WERF	7
CFA	4
RWMC	24
U.S. Highway 20	10
Lincoln Boulevard/Highway 33	13
ANL-W	12
TSF	4
LOFT	7
WRRTF	4
NRF	11
ICPP	15
TRA	13
EBR-I	1

radiation protection standard for the public (100 mrem, prolonged exposure) can be detected with the TLDs.

Surveys are also made using hand-held NaI scintillation survey meters outside of each facility fence every 3 years on a rotating schedule. The survey results are published as isopleth maps. Individual facilities also use environmental TLD packages for area monitoring and conducting surveys inside the facility fences.

An aerial radiological survey of the INEL Site was conducted during June and July of 1990 by EG&G Energy Measurements, Inc., Las Vegas, Nevada, for DOE. The survey consisted of aerial measurements of both natural and manmade radiation from the ground surface on and around the INEL Site. A report, to be completed in mid-1991, will describe the isotopic distribution of radioactivity in the survey area. A similar survey was conducted in June 1982 (EG&G, 1984).

E.4 HISTORY OF THE ENVIRONMENTAL SURVEILLANCE PROGRAM

This section presents a brief history of the environmental surveillance program in the same order as the previous section. Only the most important changes in the environmental surveillance program are addressed. Emphasis is placed on the changes in air sampling and milk sampling, the two most important pathways.

E.4.1 Air Sampling

The locations of the low-volume air samplers from 1952 through 1990 are shown in Figure E-6. The samplers have operated at about 1 to 2 ft^3/m since 1952, although the filter material, analyses, and methods have changed significantly. Statistical analysis of the air sampling results were first reported in 1975. Regular analysis for specific gamma emitters began in about 1970, although any filters with a high gross count were analyzed for specific isotopes before then. A 256-channel analyzer was installed to determine specific gamma emitters in mid-1958. By 1971, a lithium-drifted germanium (GeLi) detector was calibrated for gamma spectrometry. The GeLi detector offered superior resolution of the photopeaks over the NaI detectors. A lower detection limit was also obtained.

The present method of laboratory analysis for the weekly particulate filters was started in 1967, with analyses for gross alpha, gross beta, and iodine. This method overlapped the telemeter system that was in use from 1960 to 1970. The telemeter system is described in the 1961 Health and Safety Division report (AEC, 1962a, p. 165). The filter paper from an air sampler was cycled to a Geiger-Mueller (GM) tube once a day. One GM tube measured the buildup of particulate activity over the day and another measured the decay of the activity collected the previous day. A carbon cartridge, through which the filtered air passed, was measured with a scintillation counter. Ambient radiation was measured with an ion chamber. Results were reported each hour. Gross beta measurements were made on the filters from the telemeter system each week starting in the second quarter of 1960 through April 1968.

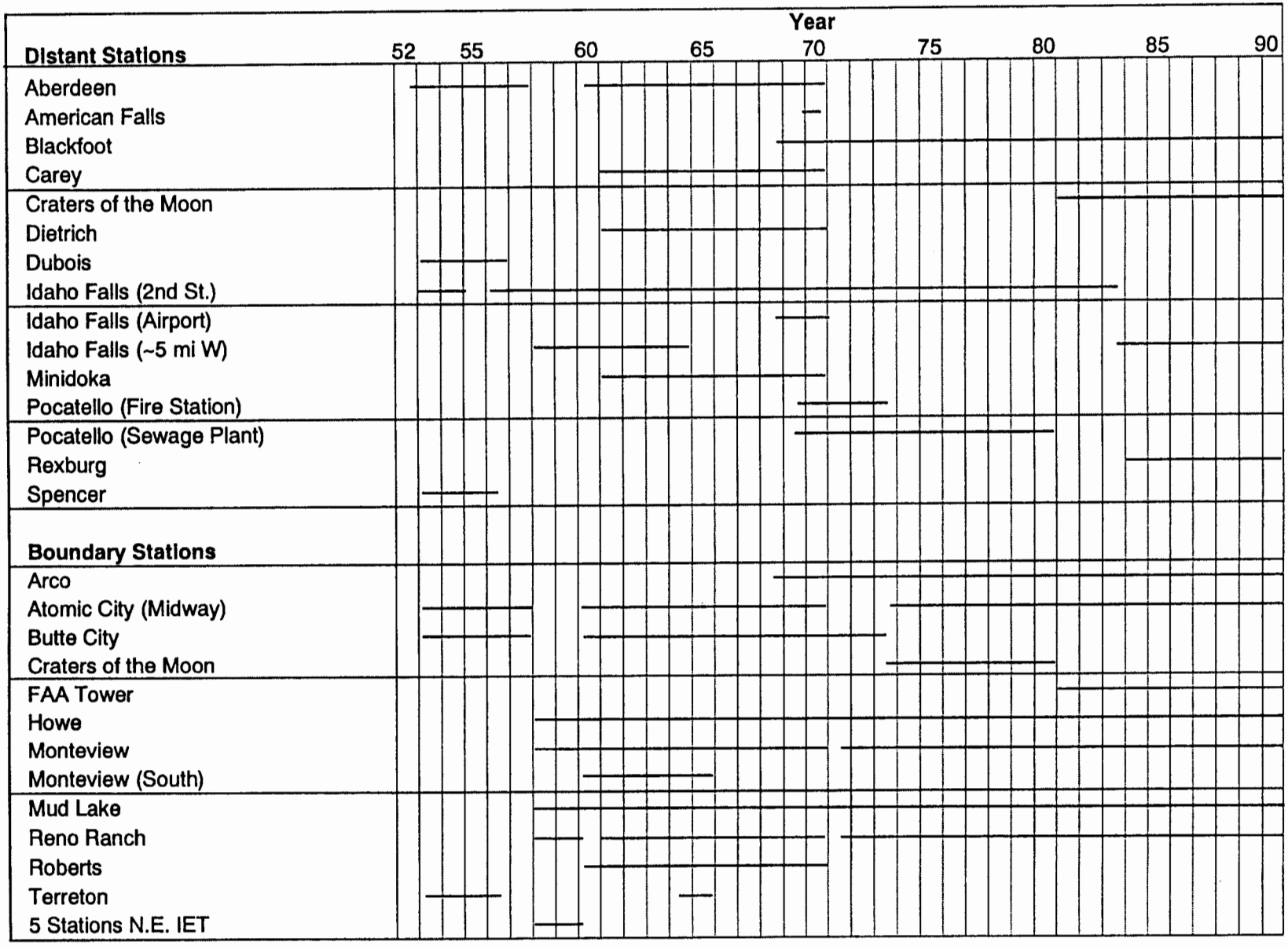


Figure E-6. Location of low-volume air samplers (1952-1990).

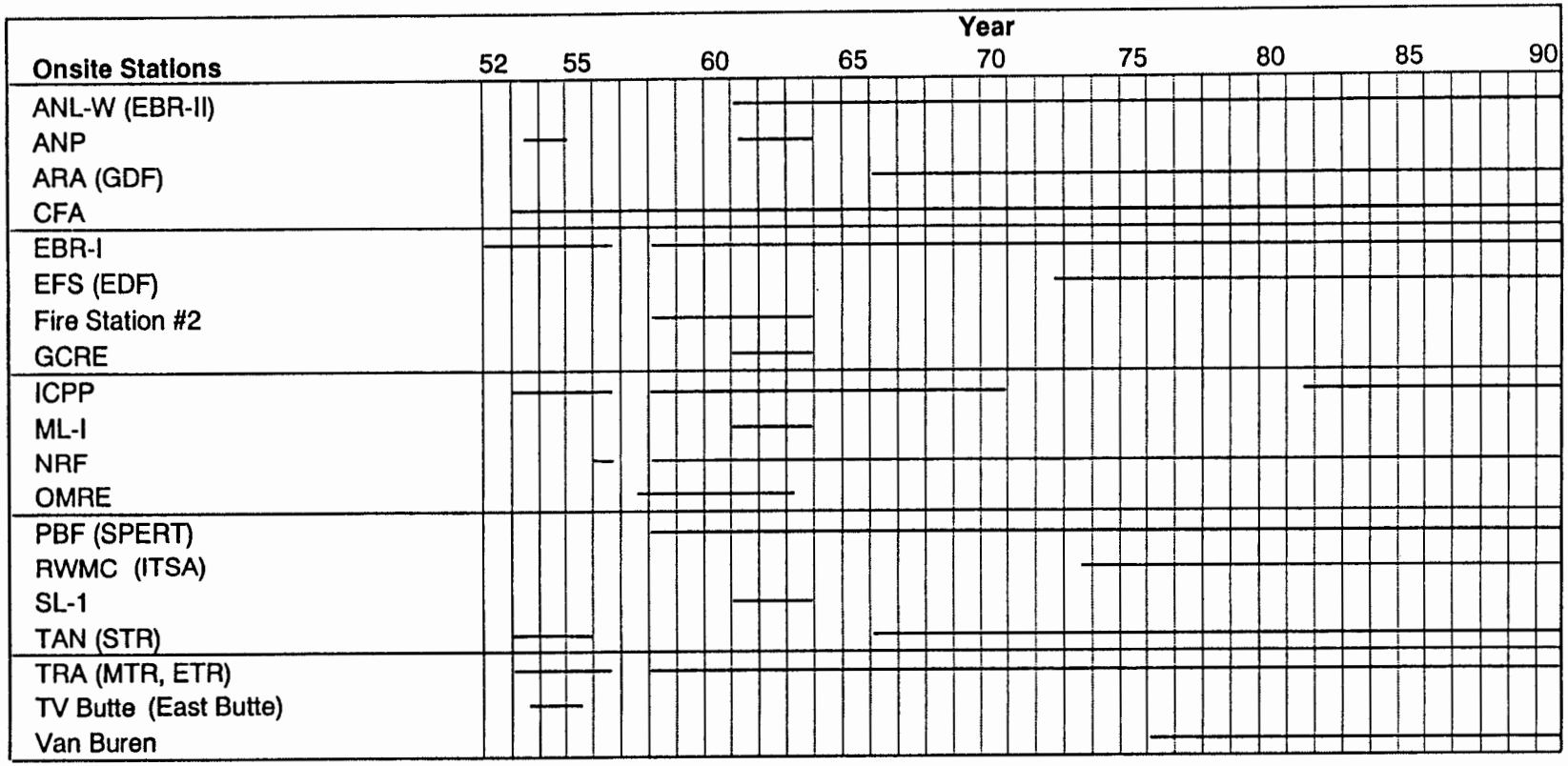


Figure E-6. (continued).

Gross alpha measurements were made on the filters in the laboratory in late 1966 and in 1967. Summary reports indicate that 6 onsite and 12 offsite telemetry stations continued in operation at least through 1970. In 1971, a new telemetry system was phased in that provided continuous meteorological data, a reading from a bare GM tube, and an air sampler that could be activated in emergencies.

Carbon cartridges, which collected radioactive iodine, were used on separate air samplers from 1958 to 1966. Through 1964, the cartridges from eight stations were counted in the laboratory, with the assumption that all of the activity was due to I-131. By 1966, the number of air samplers with carbon cartridges that operated for the full year had decreased to three.

A "continuous radiation monitoring and telemetering network" of 11 primary and 12 secondary stations had been established in 1958 and operated through 1960. The stations gave "an indication of whether the radiation level was above or below a preset level" (AEC, 1959). Only summary data were found for this period. The secondary stations were moved periodically and were activated for special tests.

From 1952 to 1957, air samplers and fallout collectors were used. An air sampler with a sampling rate of 2 cfm was equipped with a filter, readout device, and strip chart. The results were plotted in percentage of full scale. Radiation density (Rad Den) was a measurement of the number of radioactive particles deposited on a 3 x 4-in. horizontal fallout plate. The plates were analyzed by autoradiography--placing an unexposed photographic film on the plate, developing the film, and counting the number of light spots caused by radioactive particles on the plate. Both methods only provide relative results that are difficult to quantify.

The high-volume air samplers at EFS and CFA have been operating at least since 1973. The beginning date is uncertain. Because the filters are used primarily for rapid indication of INEL Site releases, records of the gross gamma results are not retained. Gamma scan results of the monthly composites are available from 1975 to present. High-volume air samplers have been used for specific events since at least 1958 (AEC, 1959). Two high-volume air

samplers were operated for the Nevada Test Site fallout from April 1958 to December 1959 in Idaho Falls and at CFA (AEC, 1960a). The records of the high-volume sampler operated in Idaho Falls since January 1973 for the EPA's National Air and Environmental Radiation Laboratory can be found in the series of reports from that laboratory and its predecessors.

Tritium in atmospheric moisture has been measured at Idaho Falls, EFS, and Van Buren since 1976. Tritium was measured at Idaho Falls, EFS, and at a point on Lincoln Boulevard halfway between CFA and ICPP in 1974 and 1975. In 1973, the EFS and Lincoln Boulevard locations were active. The development of the system was reported in the 1970 report (AEC, 1971a); therefore, some limited sampling may have been done before 1973.

Results of the air sampling program can be found in Section E.5.1.

E.4.2 Water Monitoring

The USGS has monitored ground water under the INEL Site since the beginning of INEL Site operations. Changes in the number of wells and the types of analyses are too lengthy to address in this appendix. Numerous reports document ground water monitoring and are available in the INEL Public Reading Room.

Sampling of onsite drinking water for radionuclide analysis began in at least 1958. Analysis for tritium started in 1961. Offsite, 255 water samples were collected in 1958. Offsite water sampling is mentioned in the 1959 annual environmental report at 28 locations (AEC, 1960a). Most of the sampling locations were in Minidoka, Jerome, and Gooding Counties. The number of samples was reduced in 1966 because of increased knowledge of the directional flow and flow rates of underground water and emphasis on sampling from populated areas near the INEL Site boundary.

Because the radioactivity in the ground water has not been measurable in drinking water offsite, the program will not be discussed further.

E.4.3 Animal Tissues

Extensive sampling of animal tissue, primarily as a part of radioecology research studies, was reinstated in 1973 (ERDA, 1976b; DOE, 1978b; DOE, 1983b). From 1963 to 1973, ecology research stressed the deposition of radionuclides and their movements through the food chain. From 1956 through 1965, rabbit thyroids (for I-131) and rabbit bones (for Sr-90) were frequently used as indicators of the extent of contamination around INEL Site facilities. A map of the levels of I-131 activity in jack rabbit thyroids around the Initial Engine Test (IET) Facility in 1958 was published in AEC (1959). Graphs and tables are given for I-131 and Sr-90 in jack rabbits for 1958 through 1962. However, data were no longer reported in 1962 "due to the uncertainty of correlation between Strontium 90 data for jack rabbits to human exposure" (AEC, 1962c). Other animals were also sampled in the program.

E.4.4 Foodstuff Monitoring

Milk sampling, especially for I-131, has been an important part of the environmental program for many years. The first sample result found in the records is dated October 30, 1957; however, samples may have been obtained earlier. The routine sampling program consisted of a weekly sample from a large processor in Idaho Falls and monthly samples from individual farms or processors around the Upper Snake River Plain. Since 1962, the number of routine samples collected per year is about 150. More milk samples were obtained when the Aircraft Nuclear Propulsion and RaLa Programs were active in the late 1950s and early 1960s, peaking at 790 samples in 1960.

The detection limit or minimum detectable concentration depends on the sample size, the time period over which the sample is counted, and the background counting rate of the detector used. As releases of I-131 have decreased, the approximate detection limits have also decreased because of better equipment and methods. A chronology of milk detection limits is shown below.

1958, 1959	The detection limit was approximately 1500 pCi/L for a 50-mL sample. The counting time was probably 5 minutes.
3/30/60	The detection limit was approximately 1000 pCi/L; increased sample size to 75 mL.
7/14/60	The detection limit was reduced to about 200 pCi/L; increased the sample size to 850-900 mL. A 5 x 5-in. NaI well counter was used.
1961	Both 5- and 10-minute counting times were used in the first part of 1961.
7/18/61	The detection limit was reduced to approximately 50 pCi/L; increased the counting time to 50 minutes.
9/3/62	The detection limit was approximately 20 pCi/L; increased the counting time to 60 minutes and the sample size to 3000 mL. Gamma scans for specific isotopes in routine milk samples started.
4/1/74	The approximate detection limit was reduced to 1 pCi/L; increased the counting time to 1000 minutes and the sample size to 3800 mL. Milk samples are passed through anion exchange resin, which is then analyzed for I-131 by gamma spectrometry.

The analysis method used until 1962 involved a gross count on a sodium iodide detector, subtraction of background, subtraction of a fixed count rate for K-40, and conversion of the net count rate to I-131 using a 33% efficiency for the detector for I-131. All of the activity not attributed to K-40 was assumed to be I-131.

Analysis for Sr-90 in milk has been performed since 1962. The number of samples analyzed has varied from the current level of about 10 samples per year to a high of 185 samples in 1967. Between 100 and 150 samples per year were analyzed for Sr-90 from 1963 to 1966. Tritium analyses of milk started in 1973. Analysis for Cs-137 on a regular basis started in 1963 and ended in 1973. Potassium-40 results are available from 1966 through 1973. A few milk samples were analyzed for I-129 for most years from 1979 to 1989.

The first map that identified wheat sampling locations was for the period of July to December 1964 (AEC, 1965a), although the concentrations

reported then are said to be similar to the results from the previous years. The 1962 annual report mentions ecology research on wheat samples (AEC, 1963a). The 1963 annual report (AEC, 1964a) gives results for six locations for two types of irrigation systems and on dry farm land. Samples have been analyzed for Sr-90 and Cs-137. Potatoes were sampled in 1976 (ERDA, 1977a). Lettuce has been sampled since 1977, starting with two locations in that year. Five to eight locations for lettuce have been sampled since 1979.

Results of the milk sampling program are presented in Section E.5.2

E.4.5 Soil Sampling

The present soil sampling program was outlined earlier, along with the dates that the offsite sampling became a regular program. The onsite soil sampling program has rotated among the major INEL facilities since 1973. Most soil samples collected before 1973 were analyzed for gross gamma. Soil sample results specific for Cs-137 and Ce-144 were found for as early as 1960 (AEC, 1961a). Comparisons of measured and calculated soil concentrations are made in Section E.5.3.

E.4.6 Direct Radiation Exposure

The number of locations monitored for direct radiation exposure has been about 145 since late 1968. The highest number of locations noted in reports was 290 in 1961, dropping to about 50 from 1965 through mid-1968. About 49 locations were monitored in 1959, and 33 locations were monitored in 1958. The present "card" of 5 unshielded TLDs was introduced in 1970. TLDs were also used from 1966 to 1969: two unshielded and one with a cadmium shield. Film dosimeters were used in several different cases from 1959 to 1966. The original film badge was used from 1959 to 1962. A sealed film badge was used from 1962 to 1964. A sealed film badge with a red holder, which included a desiccant, was used from 1964 to 1966. Each had a lead-shielded film and an unshielded film. Results are discussed in Section E.5.4.

Surveys with NaI scintillation meters have been conducted outside facility fences since 1973. Results were published in ERDA (1977b) and have since been published in RESL quarterly reports. Before 1973, surveys with hand-held meters had been conducted. Isopleths of radiation exposure rates near facilities are shown in the 1959 and 1960 annual reports (AEC, 1960a; AEC, 1961a).

The aerial radiological survey of the INEL Site in June 1982 was mentioned. Similar surveys were also performed by EG&G, Inc. in 1974 (Jobst, 1976) and in early 1966. An aerial survey was performed by the Health and Safety Division in 1959. The map of that survey is published in AEC (1960a).

E.5 COMPARISON OF ENVIRONMENTAL MONITORING WITH DOSE EVALUATION RESULTS

The following section will address those areas where environmental measurements can be compared to calculated concentrations or doses. The best comparisons are those measurements from air sampling data (especially for I-131), for milk sampling data, and for direct radiation measurements. In general, the environmental data can only be used to show that the dose calculations are reasonable or that the actual doses received could be no higher than a value calculated from the detection limit. Even then, the environmental data can only be compared to small parts of the total dose calculation. Some qualifications regarding the comparison of measurements to calculations are given in Section E.5.2.4.

E.5.1 Air Sampling Results

In recent years, Sb-125 was the only radionuclide released from the INEL Site in large enough quantities to be consistently measured in the offsite environmental samples. Antimony-125 was released from the ICPP in the last quarter of 1980 and in the last quarter of 1986 through July 1988. Table E-4 shows some of the measured Sb-125 concentrations in air at the boundary stations in 1986, 1987, and 1988. Also shown are the calculated air concentrations for 1987 and parts of 1986 and 1988. The calculated concentration is within a factor of five of the measured concentration. The calculated concentration is higher than the measured concentration for two-thirds of the measurements. At 20 to 30 km from the source, the agreement is considered good. There are several reasons why the measured data may not always agree with calculated concentrations: (a) concentrations at the offsite locations are low, and at those levels, the uncertainty in the measurement is approximately equal to the measurement itself; (b) the models imperfectly represent real conditions over large areas; and (c) the models assume a continuous, uniform release rate for an entire year, which is rarely the case for Sb-125.

Table E-4. Antimony-125 measured and calculated concentrations on the INEL Site boundary

Year	Location	Measured Concentration (E-15 μCi/mL)	Calculated Concentration (E-15 μCi/mL)	Ratio M/C[a]
1986 (3 months)	Atomic City	12.0	6.0	2.0
1987	Arco	2.6	0.6	4.3
	Atomic City	2.2	9.2	0.24
	Howe	4.8	6.0	0.8
	Mud Lake	2.4	6.0	0.4
	Reno Ranch	1.9	2.0	0.95
1988 (6 months)	Howe	2.9	3.6	0.81
	Atomic City	2.2	11.0	0.20
	Reno Ranch	2.5	1.3	1.9

a. Ratio of measured to calculated concentrations.

Antimony-125 was not detected at all boundary locations; only 35% of the quarterly air filter results for the tabulated periods were considered as having detectable Sb-125. Antimony-125 was detected in 7% of the distant quarterly samples in this same period.

In the period from 1980 to 1989, two international events were easily detected at the INEL: (1) the People's Republic of China weapons test of October 1980 and (2) the accident at Chernobyl in April 1986. The Sb-125 concentrations from INEL releases in the fourth quarter of 1980 were identified even in the presence of other radionuclides from weapons testing.

The concept of comparing boundary to distant locations for specific radionuclides began to develop in 1972. More boundary stations had been added in 1971, and the number of boundary locations analyzed for specific radionuclides was gradually increased from two in 1971 to seven in 1975. The number of distant stations analyzed for specific radionuclides increased from one in 1971 to three in 1975, enabling better statistical comparisons. From 1973 to 1979, the concentrations of specific radionuclides detected at the INEL Site boundary were no different from those detected at the distant

stations with the following exceptions: 1973, Nb-95 at Arco and Mud Lake; 1977, Sb-125 for the boundary group; and 1978, Cs-137 at Reno Ranch and Am-241 at Monteview. Although no statistical tests were performed, small differences between boundary and distant locations were also noted in 1971 and 1972. Fallout from nuclear weapons testing by the People's Republic of China was clearly detected in January 1972, June-July 1973, October 1976, November 1976, September 1977, and March 1978.

Gross beta analyses of air filters have been performed since 1960. Data comparing groups of onsite and offsite stations began to be plotted in 1969, with the boundary group added in 1971. The monthly concentrations of radionuclides have decreased from about 1E-12 μCi/mL in June 1969 to about 3E-14 μCi/mL in 1989, which reflects the decreased atmospheric inventory because of the reduction in atmospheric nuclear testing. In general, the differences in the monthly average concentrations of the three groups show little difference. Sharp rises in all three groups coincide with foreign nuclear atmospheric weapons tests. Differences between onsite and distant locations are shown by some of the data. In recent years, the largest differences are associated with the Sb-125 releases previously mentioned.

Air sampling has also been conducted in support of many of the tests at the INEL. Air sampling was extensive during the IETs. General results for 1958 through 1960 are given in the Health and Safety Division annual reports.

> The maximum radioactivity detected in air occurred during October and was observed at Roberts and Monteview, Idaho. This can probably be attributed to tests which were being conducted under controlled conditions at the ANP area at that time (AEC, 1961a).

IET #23 was conducted from September 1 through mid-October 1960. The maximum beta activity of a single air sample in October is listed as 11E-12 μc/cc (AEC, 1961a, p. 6). That concentration is similar to the particulate and iodine air concentrations calculated in this report, but the time period for the air filter (7 days) is different than the time period for the calculation (13 days over 5 weeks). The report also states, "the section was engaged in monitoring during power tests Nos. 18 through 26 at the Aircraft Nuclear Propulsion Area. Both mobile and fixed monitoring equipment were utilized"

(AEC, 1961a, p. 105). In 1959, the section monitored IETs #14, 15, 17 and 18. "No radiation attributed to any of these tests was detected beyond the boundary of the NRTS" (AEC, 1960a, p. 110). Monitoring of ground contamination was also performed after the ANP nuclear excursion on November 18, 1958 (IET #13). "No measurable exposure to personnel or the public occurred as a result of this incident" (AEC, 1959a, p. 55). However, the detailed air sampling records have not been found for this period.

The following constraint on operations was noted for IET #21 (FEET #1) and #23:

> When stack release of fission products exceeds 10 curies in any one hour period, age 10 minutes, the operating procedures requires [sic] that the dose predictions be verified for each significant diffusion weather type. During periods of steady, well defined wind directions, the ratio of predicted activity to the measured activity in the field must be between 1 and 10. During periods of light and variable wind conditions, any ratio on the conservative side is acceptable.
>
> During both IET 21 and IET 23, the verification results compared to predicted activities downwind from IET were all within the required limits (Pincock, 1960).

This constraint is derived from the general operating rules for testing at ANP (Levine, 1959, pp. 8-9) and is referred to in discussions of the earlier IETs #14 and 15 (Boone et al., 1959, pp. 21-23) and IET #11 (Evans, 1958, p. 27)

E.5.2 Milk Sampling Results

Milk sampling results from 1961 through 1958 are examined in the next four sections. In several instances, the potential dose to an infant thyroid has been estimated for purposes of comparison to the doses calculated from radionuclide release and atmospheric dispersion information. These estimates use the same assumptions regarding infant milk consumption rates and ingestion dose factors as used elsewhere in this report (Appendix C). They also assume that all milk consumed by an infant contains I-131 at the reported concentration for a period of time usually equal to 10 days. The calculations should not be interpreted as actual doses to infants.

The reported milk concentrations represent upper limits of I-131 in milk, since all are based on gross gamma counts. After subtraction of a background count rate and a count rate for K-40 (natural) activity, the remainder is assumed to be I-131. It is likely that some of the gross gamma count rate was due to radioiodines with shorter half-lives than I-131, or to other radionuclides. In addition, most results were close to the detection limit. At the detection limit, the uncertainty in the result is equal to the result itself. Another complication in the interpretation of results is the unquantified contribution from sources outside the INEL such as atmospheric nuclear weapons testing by the United States and the USSR.

Summaries of the milk sampling results are given below. Most of the results can be found in the annual environmental reports (see Section E.2). Results were only summarized in the earlier report series. For the years 1958 through 1964, the "coding sheets" used to enter the milk data into an IBM computer were consulted. Those sheets show the location from which the sample was obtained. The location is identified by range, township, and section (mi^2). Some are further identified to the sixteenth of a section. The sheets also show the net counts per sample, standard deviation of the count rate, and the reason the sample was collected. As mentioned earlier, the methods used for analysis of I-131 in milk from 1958 to 1962 were less sophisticated than those used today. The definition of what concentration is considered detectable also differs. The present "approximate minimum detectable concentration" (MDC) is equal to twice the "analytical standard deviation" (2s), where all the analytical uncertainties have been calculated (Hoff et al., 1990). See the discussion on interpretation of results near the MDC in Appendix C of Hoff et al. (1990) for more information. From 1958 through 1962, an arbitrary designation of a sample concentration as "significant" was made when the concentration was three times the standard deviation (σ) of the net count rate (the definition of the approximate "detection limit"). With the higher uncertainties in the analyses for these samples, the 3σ definition is appropriate. The improvement in the detection limit or MDC through time is given in Section E.4.4.

E.5.2.1 1964-1989 Milk Sampling Results

Milk sampling results for I-131 are summarized in Table E-5 for 1964 through 1989. The year 1964 was chosen as a dividing point because I-131 releases decreased substantially in that year when the RaLa processing ended. The IET releases ended in 1961. As noted in the table, many of the positive results are attributed to atmospheric weapons testing or to the Chernobyl

Table E-5. Milk sampling results for I-131 (1964-1989)

Year	Offsite Samples	Samples with Detectable I-131	Source of I-131 (Number of Samples)			
			INEL	Weapons Test	Chernobyl Accident	Undetermined Causes
1964	160	0				
1965	144	10		8		2
1966	139	24		8		16
1967	185	6		6		
1968	185	0				
1969	185	0				
1970	172	0				
1971	172	3				3
1972	172	0				
1973	153	0				
1974[a]	153	11	1	10		
1975	152	8				8
1976	156	8		8		
1977	156	23		23		
1978	152	3		2		1
1979	146	0				
1980	148	9		5		4
1981	147	2				2
1982	155	2				2
1983	154	8				8
1984	157	0				
1985	172	0				
1986	170	39			37	2
1987	146	0				
1988	142	0				
1989	144	0				

a. The minimum detectable concentration was lowered from 20 to 1 pCi/L in April 1974.

accident. Of the remainder, most are close to the MDC and may not reflect actual concentrations.

Of the samples identified as having detectable I-131 in the period from 1974 to 1989, not including those attributed to weapons testing or Chernobyl, 18 were less than the approximate MDC of 1 pCi/L. They were listed as detected because an evaluation of the uncertainty of each result indicated that the result met the criteria for detection. Another eight samples measured 1.0 to 1.8 pCi/L. Two samples measured 4 $\pm$ 2 and 4 $\pm$ 3 pCi/L; the higher uncertainty indicates an unusually long delay between sample collection and analysis. The three detectable concentrations (21, 22, and 23 pCi/L) in 1971 were just over the minimum detectable concentration of 20 pCi/L. The 16 detectable concentrations in 1966 were clustered from May 22 to July 24, mostly from Idaho Falls. The cause was not indicated in the reports and was not found in this review. No details are available for the two 1965 detectable results that were not related to weapons testing. Sample results attributable to weapons testing and Chernobyl varied considerably, with a few samples in the hundreds of pCi/L range.

E.5.2.2 1963 Milk Sampling Results

In 1963, 25 curies of I-131 were reportedly released from the INEL. This was due to four RaLa runs in January and February 1963. None of the 199 milk samples for 1963 were above the detection limit for I-131 of 20 pCi/L. If the milk concentration were equal to the minimum detection limit for the entire 2 months, an infant thyroid dose of *less than* 15 mrem can be calculated. This represents an upper bound based on the possible I-131 concentration in the collected samples. Using the methodology described in Appendix C, the infant thyroid dose was calculated to be 3.4 mrem based on the release of 25 Ci in 1963. Thus, milk sampling results, although limited, tend to substantiate the calculated dose.

E.5.2.3 1962 Milk Sampling Results

In 1962, 123 of 259 raw milk samples contained detectable I-131. One report says "113 milk samples were collected on a monthly basis from 10

stations" (AEC, 1963a, p. 82). The difference in the reported numbers of samples is probably a result of not including "background" samples in the total number. The code for background was noted on 73 of the 259 samples, bringing the numbers into better agreement. Some Idaho Falls samples also have the monthly code noted. Many (54 of 82) of the detectable samples were collected from June 10 to September 3. In September, the detection limit was lowered from 50 to 20 pCi/L. The analysis was also made specific for I-131 by using gamma spectrometry rather than gross gamma. After September 3, 60 of 106 samples had detectable I-131 concentrations. However, for all of these samples, there is no clear pattern of decreasing concentration with distance or direction from the INEL Site. Some 35 of the 60 detectable results were from background areas, most from Idaho Falls. Four boundary concentrations were higher than the average background concentrations and are identified in Table E-6. These higher concentrations do not coincide with the RaLa releases. The major I-131 releases from RaLa were associated with extensive decontamination operations that started on November 27, 1962. The SPERT-I, Test No. 1 was conducted on November 5, 1962, but the reports state that no halogens were detected, even with intensive monitoring close to the point of release. The following excerpt also indicates the radioactivity in milk was not INEL Site related.

> Strontium-90 in milk produced near the NRTS during 1962 reached an average high in July of 13 picocuries per liter, after which it declined to near February levels. It was noted that milk produced from last year's hay remained at low levels of 2 to 5 picocuries per liter all year long. At the time of the strontium-90 high point in mid-year, iodine-131 was readily detected in milk, averaging 160 picocuries per liter, but dropped significantly for

Table E-6. I-131 boundary concentrations

Date	Location	Concentration (pCi/L)
9/16/62	Howe	200
10/14/62	Arco	140
11/14/62	Arco	320
11/14/62	Monteview	200

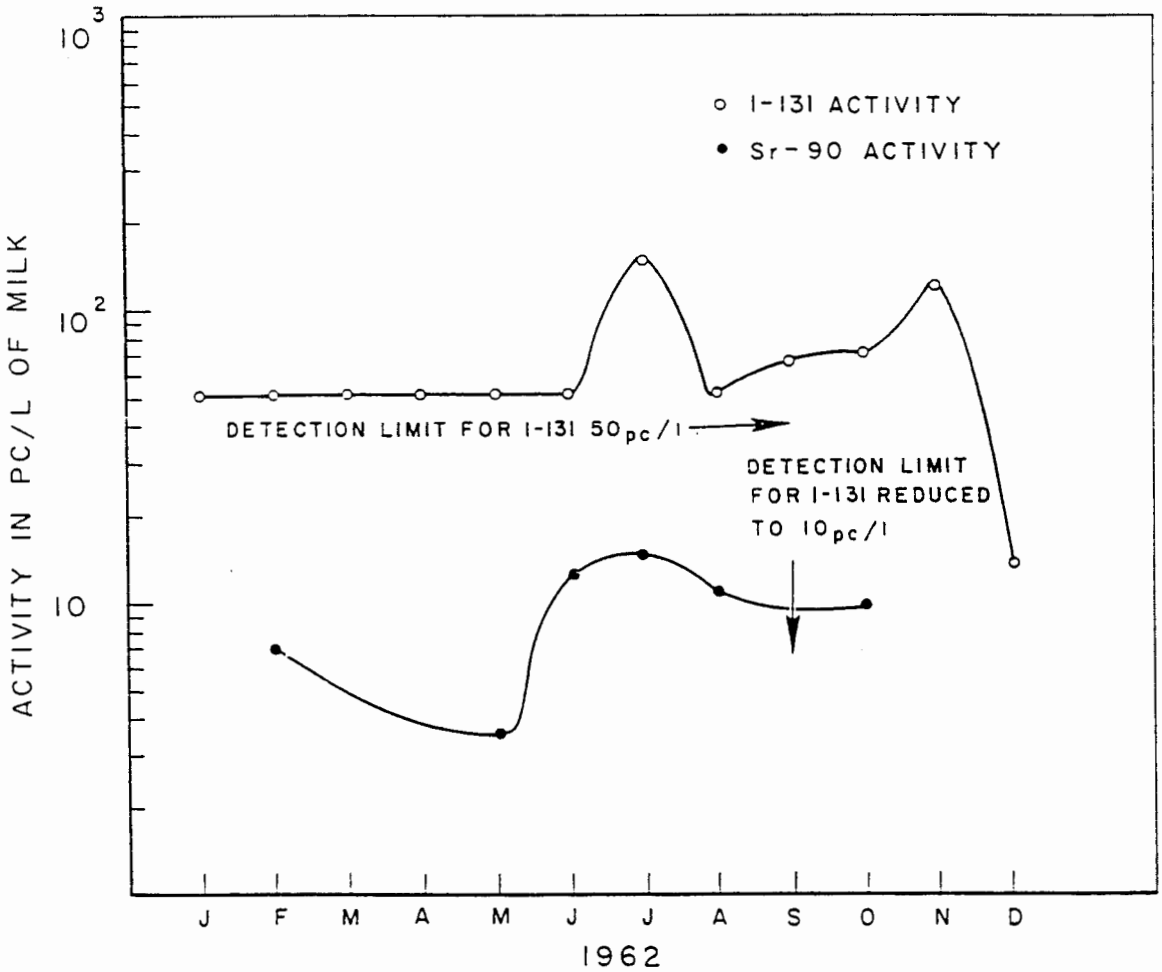
> the remainder of the year (Figure 13) [note: See Figure E-7]. On the basis of the strontium-90 and iodine-131 analyses, at no time during 1962 was the NRTS generated radioactivity detected in offsite milk or jack rabbit bones (AEC, 1963a, p. 56).

One possible explanation for some of the detectable results is the Sedan cratering test at the Nevada Test Site on July 6, 1962. Undoubtedly, foreign nuclear tests were conducted because the Limited Test Ban Treaty did not go into effect until August 5, 1963. "As was expected due to international nuclear tests, an increase of iodine in milk was noted throughout the country" (AEC, 1963a, p. 82).

E.5.2.4 1961 Milk Sampling Results

Milk sampling results from 1961 through 1958 are examined in the next four sections. Infant thyroid doses are calculated from reported milk concentrations for comparative purposes only. The calculations should not be interpreted as actual doses to infants. The reported milk concentrations represent upper limits of I-131 in milk because all are based on gross gamma counts. After subtracting the background count rate and a count rate for K-40 (natural) activity, the remainder is assumed to be I-131. It is likely that some of the gross gamma count rate was due to radioiodines with shorter half-lives than I-131 or to other radionuclides. In addition, most results were close to the detection limit. At the detection limit, the uncertainty in the result is equal to the result itself. Another complication in interpreting results is the unquantified contribution from atmospheric nuclear weapons testing by the United States and the USSR.

The detectable I-131 concentrations in milk in 1961 were associated with the SL-1 Accident and with foreign nuclear testing. Six of seventy-eight samples collected specifically for the SL-1 accident contained detectable I-131. Table XLIII on page 134 of AEC (1962a) shows the dates, locations, and concentrations.



Reproduced from AEC (1963a)

Figure E-7. Average I-131 and Sr-90 found in raw milk.

> Milk sampling was started on January 4. Between January 4 and 19, 28 milk samples were collected from five farms in the Taber area (Figure 52). One farm was located at Taber and the remaining four farms were eight miles south of Taber. These farms with dairy cows were located nearest to the southern boundary of the NRTS. The results of the milk-sampling program through January 19 are presented in Table XLIII. Six of the 28 samples showed gross gamma activity slightly greater than three standard deviations of the net count. The remainder of the samples showed no detectable activity under the conditions used for counting, which were capable of detecting 2×10^{-7} microcuries of iodine-131 per milliliter.
>
> The results of the milk-sampling program confirmed that any iodine-131 present in milk, as a result of the SL-1 incident, was well below the Radiation Concentration Guide of 2×10^{-6} microcuries per milliliter for non-occupational exposure. Low iodine-131 levels in milk were expected because dairy cows are fed baled hay and are not on pasture during this time of the year (AEC, 1962a, p. 134).

Four milk samples obtained from Taber (10 mi southeast of Atomic City) in a 10-day period averaged 200 pCi/L. The calculated infant thyroid dose from consumption of milk of a concentration of 200 pCi/L for 10 days is 25 mrem. The infant thyroid dose calculated in this report for the SL-1 Accident was 173 mrem. The estimate for Atomic City is clearly conservative because there were no milk cows at that location (AEC, 1962a, p. 134).

None of 91 samples showed detectable I-131 from February 10 to August 17, 1961. In September, October, and November 1961, 30 of 66 samples were identified as having detectable I-131. These data were attributed to the foreign atmospheric nuclear testing. A graph on page 174 of AEC (1962a) shows numerous Russian tests in this time period. Most of the detectable samples were from the Idaho Falls-Blackfoot area, indicating widespread fallout not connected with the INEL Site. There were no INEL tests or experiments at this time and no unusually large releases because of the RaLa Program. Two of 19 December results were just above the detection limit.

> Milk samples were collected monthly for iodine-131 analysis from the 12 perimeter farms shown in Figure 51. Milk samples of 900 milliliters were screened for iodine-131 activity in a gamma well scintillation counter. Detection methods were improved during the second half of the year which lowered the detection limit from 2.0×10^{-7} to 5.0×10^{-8} microcuries per milliliter. To lower the detection limit the counting time was increased from 10

to 50 minutes. The highest concentrations of iodine-131 observed in raw milk occurred during September in samples collected from cows on pasture. This activity is attributed to the atmospheric weapons tests being conducted by the Russians (AEC, 1962a, pp. 129-130).

E.5.2.5 1960 Milk Sampling Results

Through August 24, 1960, almost none of the milk samples contained detectable I-131 concentrations. Of the 235 samples collected, only two were marked as significant. The two samples, obtained in early January and late March, were 1400 and 700 pCi/L, respectively--near the detection limits. The notation on the samples indicates they were taken because of the RaLa processing. Neither of the detectable samples correspond to any of the higher RaLa releases for late 1959 and early 1960. The infant thyroid dose calculated from the higher concentration, if maintained for 10 days, would be 170 mrem. The 1960 infant thyroid dose from all operational releases was calculated in this report to be 20 mrem. The two detectable samples appear to be due to the counting statistics and not true concentrations of I-131 in milk because (a) no other milk samples from the same area were detectable, (b) the 1400 pCi/L sample was collected in early January when the cows would not be on pasture, (c) the dose calculations from the milk results do not agree with the dose calculations from the releases, and (d) the samples were near the detection limit.

The sampling from August 25 through November 3, 1960, does provide information on the IET releases. The summary of the sampling performed for four of the tests is taken from AEC (1961a, p. 92) and is shown in Table E-7.

Table E-7. Summary of the 1960 IET milk sampling program

Test	Sampling Period	Farms Sampled	Samples Collected	Samples with I-131, I-133	Farms with Activity
IET #21 (FEET #1)	7/20-8/24	29	45	0	0
IET #22 (LIME)	8/25-9/1	17	200	24	12
IET #23 (FEET #2)	10/13-10/21	18	120	19	9
IET #24 (LIME-II)	10/25-11/3	15	145	0	0

An average concentration of I-131 in milk from 13 samples from the highest location in the Monteview area for the IET #22 release was calculated to be 188 pCi/L for a 7-day period. Because the average includes seven samples where the concentration was below the detection limit, the average is less than the detection limit. Currently, good practice requires averaging all samples, not just those that are detectable, which would give a biased result. The calculated infant thyroid dose from this concentration in milk for 7 days would be 16 mrem. The infant thyroid I-131 ingestion dose calculated in this report from the release information was 0.6 mrem.

For the IET #23 release, an average concentration of I-131 in milk from three samples from the highest location in the Monteview area was calculated to be 400 pCi/L for a 10-day period. The three detectable samples were spread over a 6-day period with no nondetectable samples between them. The calculated infant thyroid dose from this concentration in milk for 10 days would be 50 mrem. The infant thyroid I-131 ingestion dose calculated in this report from the release information was 4 mrem.

For the IET #24 (LIME-II) release of October 26, 1960, AEC (1961a) reported no samples were identified as having detectable I-131 concentrations. The milk sampling records show seven samples that were marked as significant. Two samples were taken on October 27 and five samples were taken on October 28. The I-131 concentrations in milk do not peak for 3 to 5 days after deposition. Over 100 samples were collected in the next 5 days and none showed detectable I-131. No explanation is given in the reports, but the detectable concentrations do not appear to be attributable to the IET #24 release. The milk detection limit at this time maintained for 10 days, corresponds to an infant thyroid dose of about 25 mrem.

E.5.2.6 1959 Milk Sampling Results

Only 5 of 808 milk samples from February 2, 1959, to September 3, 1959, were marked as significant. Each of the five results was near the detection limit. Two samples were taken at the time of IET #14, one during IET #15, and two are marked as sampling for RaLa. The calculated upper limit of infant thyroid dose based on the milk sampling for any of the episodic events is

about 170 mrem. The dates of the detectable I-131 concentrations do not correspond to the higher I-131 releases from the ICPP for 1959.

Ten samples collected on October 27, 1959, showed three detectable concentrations, and 12 samples collected on November 16, 1959, showed one detectable concentration. One of the October samples was analyzed by gamma scan, and the notation indicates that the activity detected was natural radioactivity. Both sets of samples were marked as background samples, but that designation seems questionable. The October samples could be a result of the 1959 ICPP Criticality, and the November samples could be a result of IET #17. All samples, except the sample with natural radioactivity, had concentrations near the detection limit.

On December 14 and 15, 1959, 47 samples were collected in support of IET #17. Sixteen of these samples were identified as significant. The two highest samples were gamma scanned and had only natural radioactivity. Of the remainder, the highest concentration (if maintained for 10 days) at any single location would result in an infant thyroid dose of about 180 mrem. The calculated dose for the infant thyroid from I-131 for the IET #17(B) release was 0.3 mrem. Note that this test was conducted during the winter when no pasture grass would be growing. It is difficult to postulate a reasonable mechanism that would produce the concentrations recorded in the milk at this time of year.

There is some discrepancy between the milk data from the coding sheets and the following statement from the annual report.

> Iodine-131 in milk: The milk monitoring program was confined to farms in the Mud Lake and Monteview areas NE of the Site boundary. Collections of 12 one-quart milk samples were generally made a few days after each RaLa release and at the completion of each IET operation. Milk from the individual farm cans was obtained at two creameries at Rexburg and at Ririe: Nineteen collections were made during the year. Each of the two December collections consisted of samples from 24 farms. A total of 237 samples were collected and analyzed during the year for I-131. Fifty farms were sampled during the year. A 50 ml subsample from each quart sample was analyzed. The detection limit for I-131 was 9×10^{-7} μc/ml of milk. I-131 was not detected in any of the 237 milk samples collected during 1959 (AEC, 1960a, p. 95).

There is a slight discrepancy in the number of samples--237 compared to 249. The detection limit for this year is given elsewhere as 1.5E-6 μc/mL of milk. The later figure is more consistent with the data. The statement that is hard to reconcile is that "I-131 was not detected in any of the 237 milk samples." It may be that (a) the assumption was made that because the highest concentrations in the December 14 and 15 samples were due to natural radioactivity, all of the detectable concentrations in that batch of samples were due to natural radioactivity; (b) an equipment, sampling, or analysis error was found later; or (c) the December results were just missed when the annual report was written. Without some reason to the contrary, the individual milk results shown above were interpreted as indicating detectable concentrations of I-131.

E.5.2.7 1958 Milk Sampling Results

For the period from December 30, 1957, through February 24, 1958, about 150 samples were collected. About 130 of these samples were identified as having been collected for the IET operations and 19 for the RaLa operations. Only one sample was marked as significant, and it was near the detection limit. Iodine-131 was not detected in a recount of the original sample.

On February 25 and 27, 1958, 6 of 23 samples are marked as significant. Four of these are recounts or splits from the same location. Splits or recounts of two of the samples did not confirm the presence of I-131. Thus, only one location had a detectable concentration of I-131. The average of seven counts at that location over a 6-day period was 980 pCi/L. The milk sampling corresponds to part of IET #10. If this concentration were maintained for 10 days, the calculated infant thyroid dose would be about 120 mrem. The calculated infant thyroid dose in this report for the IET #10(A) release was 5 mrem.

From February 28, 1958, through March 11, 1958, 82 samples were collected. Two of the samples were taken from Howe, Idaho. Three were marked as significant, including one of the samples from Howe. The highest was 1.5 times the detection limit. There are too few samples to estimate a dose.

IET #10(B) was conducted from March 1 to 3, 1958. The infant thyroid dose calculated in this report for the IET #10(A) release was 15 mrem.

From March 17 through May 2, 1958, 123 samples were collected. One hundred and four samples were taken in support of the RaLa processing, and 20 samples were taken for background. Five results on April 6 and 7, 1958 were marked as significant. One was marked in error. One result was a split or recount. Two results were not substantiated by recounts. Thus, only one location showed detectable I-131 near the detection limit and a dose was not calculated. It is more likely that the activity was due to the IET #11 release rather than RaLa processing because of the timing of the results. The infant thyroid dose from IET #11 was calculated in this report to be 4 mrem.

From May 3 through 6, 1958, 83 samples were collected for IET #12 (BOOT). Only one was marked as significant, and that was not confirmed by a recount.

From May 29 through June 26, 1958, 55 samples were collected to monitor the RaLa processing. Fifteen of those samples were marked significant. Twelve of those were collected on May 29 and 30, 1958. Three of the higher I-131 releases from the RaLa processing occurred in May and early June. Two samples from the highest location averaged 1780 pCi/L. That location did not have detectable concentrations of I-131 on June 2, 1958. If 30 days is assumed as the exposure period, the infant thyroid dose would be about 650 mrem. The annual infant thyroid dose from I-131 operational releases was calculated in this report to be 640 mrem for 1958. The May-early June RaLa releases comprise about one half of the 1958 I-131 releases. Thus, considering the uncertainty in the exposure period, the two calculations are in good agreement.

Of the eight samples collected for background on August 12, 1958, one from Menan, Idaho, had detectable I-131. Sampling for the Fission Product Field Release Test (FPFRT) releases was conducted from September 2 through 22, 1958. Two of the sixty samples contained detectable concentrations of I-131 near the detection limit. On October 2 and 3, 1958, 24 samples were collected

to monitor the RaLa processing. Two were marked as significant. Both were near the detection limit. There are too few data to draw any conclusions.

The samples collected from October 23 through November 3, 1959, are designated as RaLa samples. Of the 117 samples, 45 were listed as having detectable concentrations. However, the October 1958 period is known to have had high fallout from atmospheric weapons testing, and it is confirmed by the air sampling results for the Weapons Test Fallout Program. A graph is shown on page 119 of AEC (1960a). On page 57 of AEC (1959) the notation is made that the autoradiographs of the sticky paper at Idaho Falls for October 16, 25, 26, 27, and 28 were saturated--particles were too numerous to identify individually. Those dates agree well with the dates of the detectable milk samples. In addition, on October 28, 29, and 30, many of the milk samples were obtained from a location near Idaho Falls and had similar I-131 concentrations. The highest single location was near Hamer, Idaho. The average concentration of six samples was about 5600 pCi/L. The infant thyroid dose from exposure to these fallout concentrations would be about 680 mrem. The October samples are attributed to weapons testing and not to INEL operations.

Four more samples collected in November did not contain detectable I-131. It should be noted that goats purchased by the Health and Safety Division were used as monitors for I-131 in the environment. Most of the samples of goat's milk are from an area near the chemical processing plant. For the IET #12 (BOOT) release, two goats were used. "At a location approximately eight miles from the source, I-131 activity was detected in the May 4th and 5th milk of one goat. The other animal refused to eat at the time" (AEC, 1959).

E.5.2.8 Conclusions from Milk Sampling Results

The milk sampling results are not inconsistent with the doses from I-131 calculated in this report. Doses estimated directly from milk sampling results have large uncertainties because measured concentrations were often close to the detection limit. The detection limit of the analyses for I-131 in milk was dramatically lowered in the early 1960s. However, the quantities

of I-131 released also decreased substantially in the same period. The early (1958-1962) results were not specific for I-131 and have greater uncertainties than indicated simply by the counting error. The results may also overstate the I-131 concentrations because of the presence of other radioiodines and some Cs-137 in the milk samples.

E.5.3 Soil Sampling Results

Since the start of the distant and boundary soil sampling program in 1970, the two groups of samples have shown no statistical differences attributable to the INEL operations. The results are included in the annual reports, generally in even-numbered years. The results of the onsite soil sampling program have been documented, but no INEL onsite soil contamination (as distinguished from regional weapons testing fallout radioactivity) is closer than 2 mi to any INEL Site boundary or closer than approximately 5 mi to any occupied residence.

A calculation was performed to determine if the Sr-90 and Cs-137 concentrations measured at Atomic City, the location most frequently appearing as the location of the maximally exposed individual, were consistent with the deposition of Sr-90 and Cs-137 calculated in this report. The average concentrations of Sr-90 and Cs-137 in offsite soil samples in 1970 through 1975 were 40 and 54 nCi/m^2, respectively, in the top 5 cm of soil. The average soil concentrations at Atomic City (1971-1975) for Sr-90 and Cs-137 were 37 and 54 nCi/m^2, respectively. The distant concentrations were essentially equal to the concentrations at Atomic City. The uncertainty in the average distant concentrations is $\pm$20%.

The amounts of Sr-90 and Cs-137 deposited at Atomic City from operational releases before 1971 were calculated to be 10 and 7 nCi/m^2, respectively, as decayed to 1971. The year 1970 was chosen to be the cutoff because more than 90% of the radioactivity was deposited before the early 1970s. Since 1971, the radioactive decay of Sr-90 and Cs-137 in measured soil concentrations is noticeable. Thus, the amounts of these two radionuclides deposited at Atomic City from INEL operations are about equal to the amounts

that could have gone undetected in the presence of fallout from atmospheric nuclear weapons testing.

E.5.4 Direct Radiation Results

The measurements of direct radiation since 1973 indicate no statistically significant differences between boundary and distant locations. The 1970, 1971, and 1972 reports state that there is "no evidence that any of the offsite exposures were the result of NRTS activities" (AEC, 1971b; AEC, 1972b; AEC 1973). The TLD system in place from 1970 to the present will easily measure the external radiation component of background of approximately 120 mR per year. The uncertainty in the measurement is approximately 5 mR for each 6-month monitoring period. However, the seasonal variation because of snow cover on the ground surface and the location variation because of differing natural radionuclides in the local soil is about ±10 mR per year. Direct radiation exposures because of INEL operations from 1970 to the present have been well below the uncertainty in the measurement and the variation in natural background.

Before 1970, a badge with one or two TLD chips was used to measure direct gamma radiation. The uncertainty in the measurement by this method was about ±10 mrem per period that the badge was in the field. In 1967, 1968, and 1969, the badges were changed semiannually. The maximum radiation exposures measured in 1967 and 1968 were 160 and 145 mrem, respectively (1967 measurements included weapons test fallout) (AEC, 1968a; AEC, 1969b). These measurements are slightly above the current estimates of background. In 1969, the maximum perimeter radiation exposure was 15 mrem per 6 month above natural background (AEC, 1970b). The estimated effective dose equivalent (EDE) in this report for 1969 from external exposure pathways was less than 1 mrem.

In 1966 and before, film badges were used to measure direct gamma radiation. The uncertainty was again about ±10 mrem per period. Badges were changed monthly, every 6 weeks, or quarterly. The uncertainty of the measurement for a monthly badge change can be equal to the measurement because of natural background of 10 mrem per month. For 1962 through 1966, the offsite stations show "zero" radiation exposure except as noted below. Any

reading of less than 10 mrem per period was probably recorded as zero. For the period from June 29, 1964, to September 29, 1964, 12 of 13 locations show gamma exposure of 10 to 65 mrem. The highest exposure was at Mud Lake, but Idaho Falls and Carey, both distant sampling locations, measured 30 and 55 mrem, respectively. No pattern seems to exist in locations with positive results; therefore, the readings do not appear to be INEL Site-related. Three distant stations showed 25 mrem exposures in March 1964 and were probably not INEL Site-related. From May through August 1963, a few locations showed 10 to 15 mrem gamma exposures. The RaLa processing had ended by that time, but it is possible that some contribution may be due to the INEL. From May 15, 1962 through October 1, 1962, a number of offsite locations ranged from 10 to 90 mrem (with the largest reading in Idaho Falls) gamma exposure per 6 weeks. The results were higher at the distant locations than at the perimeter locations. This is the same period in which detectable milk concentrations were attributed to atmospheric weapons testing. No logical relationship between these measurements and INEL Site activities is apparent. The calculated EDE from external exposure pathways from 1966 to 1962 ranged from 0.9 to 3 mrem. Consistent with the results from this time period, Site operations would not have been expected to produce a discernable reading above background as measured by film badges.

In 1961, "with the exception of the stations surrounding the NRTS burial ground and the SL-1, all results were not significantly above the present lower detection limits of 10 millirem" (AEC, 1962a, p. 96). The 1961 annual report (AEC, 1962a, p. 96) also mentions a moisture problem, prompting a return to a 6-week exposure time. The problem of "light leaks" reported in 1960 did not occur in 1961. "It was very apparent that the red semi-transparent plastic film badges were filtering out that portion of the solar spectrum causing the light fogging" (AEC, 1962a, p. 95). The 1960 report states, "with the exception of the stations surrounding the burial ground, all results were not significantly above the present lower detection limit of 10 mrem" (AEC, 1961a).

In conclusion, the direct radiation results are not inconsistent with the calculations in this report. The measurement method and frequency before 1967 are sufficient to show that no exposures greater than about half of the

exposure rate from natural background could have occurred because of INEL Site activities. It is unlikely that most of the readings were related to INEL Site operations because many of the positive results occurred at the distant locations and not the boundary locations around the INEL Site. In other cases, the results at distant and boundary locations were nearly the same. If the results were related to INEL Site activities, the monitoring results at boundary locations would be expected to be considerably greater than at the distant locations. Since 1967, the interpretation of results is limited by seasonal variations and the variation in background radiation exposure rates and demonstrates no discernable impact from INEL operations.

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APPENDIX F

APPLICABLE STANDARDS

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F.1 APPLICABLE STANDARDS

The radiation protection standards and regulations shown in Table F-1 are applicable at the Idaho National Engineering Laboratory (INEL) Site boundary for the last year covered by this evaluation. Those standards that apply to all exposure pathways were adopted as interim standards on August 5, 1985, by the U. S. Department of Energy (DOE) (Vaughan, 1985; Tseng, 1987). These standards follow the recommendations of the International Commission on Radiological Protection (ICRP) in Publications 26, 39, and 43 (ICRP, 1977; ICRP, 1984; ICRP, 1985) and of the National Committee on Radiation Protection (NCRP) in Report Number 91 (NCRP, 1987). The U. S. Environmental Protection Agency (EPA) issued regulations for airborne radionuclide emissions from DOE facilities on February 5, 1985 (CFR, 1985), and the regulations became effective for calendar year 1985. Because of the hydrology of the INEL Site and the absence of viable exposure modes other than from airborne emissions, the doses calculated in this report from 1985 through 1989 should be evaluated against the EPA standard for airborne emissions. Thus, from 1985 to 1989, the dose equivalent to the maximally exposed member of the public from airborne emissions of radioactivity from DOE facilities was limited to 25 mrem to the whole body and 75 mrem to any organ. While the effective dose equivalent calculated for this report is not exactly the same as the dose equivalent to the whole body, the effective dose equivalent includes all of the dose to the whole body and the contribution from specific internally deposited radionuclides in organs of the body. The effective dose equivalent therefore represents a reasonable comparison to the dose equivalent to the whole body for 1985 through 1989. In December 1989, the airborne emission standard for DOE facilities was reduced to 10 mrem/yr, effective dose equivalent. The reduction was based on EPA's assessment of dose limits that are achievable by applying the best available control technology to airborne effluents.

The emission standards apply to annual releases and not to accidents. Protective Action Guides have been established for accidents. The Protective Action Guides are numerically much higher than the emission standards and define the levels where action should be initiated following an accidental release.

Table F-1. Radiation standards for protection of the public, 1985-1989

All Pathways	Effective Dose Equivalent[a] (mrem/yr)	(mSv/yr)
Occasional annual exposures	500	5
Prolonged period of exposure[b]	100	1

Emissions to Air Only[c]	Dose Equivalent (mrem/yr)	(mSv/yr)
Whole body	25	0.25
Any organ	75	0.75

a. The effective dose equivalent for any member of the public from all routine DOE operations (natural background and medical exposures excluded) shall not exceed the values. Routine operations means normal planned operations and does not include actual or potential accidental or unplanned releases.

b. A prolonged exposure is one that lasts, or is predicted to last, longer than 5 years.

c. Limits of 40 CFR 61, Subpart H, by the EPA (CFR, 1985). In December 1989, the standard was reduced to 10 mrem/yr effective dose equivalent, beginning in calendar year 1990.

Because the calculations of dose to an offsite individual go back to 1952, we also traced the radiation protection standards back to that time. To understand the standards, one must understand a recent change in the approach to radiological protection recommended by the ICRP. The previous approach was to protect the whole body and the critical organ of the body. The radiation standards for some specific organs of the body were higher than for the whole body because of the lesser sensitivity of those organs to radiation exposure. Each dose assessment required a calculation of dose commitment to the whole body and to the critical organ, the organ most exposed relative to the corresponding standard. In 1976, the ICRP recommended a single measure of relative risk of adverse health effects, the effective dose equivalent. Under this approach, the relative risk to each organ is to be considered. The effective dose equivalent is calculated by multiplying each committed dose equivalent to a specific organ from internal sources of radiation by its weighting factor. The results are summed with the dose equivalent to the whole body from external radiation sources to produce a single dose number; the number reflects the relative risk to the various organs and the whole body combined. This report calculates an effective dose equivalent, which is a reasonable comparison to the dose equivalent to the whole body. For comparison to the EPA standards and to earlier DOE standards, the 50-year committed dose equivalent to the critical organ is also listed.

From 1981 through 1984, the radiation protection standards for the public were provided in DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection" (DOE, 1981, p. XI-5). These standards are shown in Table F-2. Note that these standards are based on an annual dose equivalent of 0.5 rem (500 mrem) to the whole body and 1.5 rem (1500 mrem) to other organs.

In October 1977 the Energy Research and Development Administration (ERDA) was combined with several other federal agencies into the DOE. The radiation protection standards applicable for the years 1977 to 1980 were those of ERDA Manual Chapter 0524, dated March 30, 1977 (ERDA, 1977); they were still in use even though the name of the organization had changed. The applicability of the ERDA Manual Chapters was stated in DOE Interim

Table F-2. Radiation protection standards for members of the public (DOE), 1981-1984[a]

Type of Exposure	Annual Dose Equivalent or Dose Commitment[b]	
	Based on Dose to Individuals at Points of Maximum Probable Exposure (rem)	Based on Average Dose to a Suitable Sample of the Exposed Population[c] (rem)
Whole body, gonads, or bone marrow	0.5	0.17
Other organs	1.5	0.5

a. DOE (1981)

b. In keeping with the DOE policy on lowest practicable exposures, exposures to the public shall be limited to as small a fraction of the respective annual dose limits as is reasonably achievable.

c. See Paragraph 5.4, FRC (1960), for discussion on concept of suitable sample of exposed population.

Table F-3. Radiation protection standards for external and internal exposure for individuals and population groups in uncontrolled areas (ERDA), 1977-1980[a]

Type of Exposure	Annual Dose Equivalent or Dose Commitment[b]	
	Based on Dose to Individuals at Points of Maximum Probable Exposure (rem)	Based on Average Dose to a Suitable Sample of the Exposed Population[c] (rem)
Whole body, gonads, or bone marrow	0.5	0.17
Other organs	1.5	0.5

a. ERDA (1977)

b. In keeping with ERDA policy on lowest practicable exposures, as expressed in Chapter 0524-011b, exposures to the public shall be limited to as small a fraction of the respective annual dose limits as is practicable.

c. See Paragraph 5.4, FRC (1960), for discussion on concept of suitable sample of exposed population.

Management Directive No. 5001 (DOE, 1977). The radiation protection standards of ERDA Manual Chapter 0524, *Standards for Radiation Protection*, are shown in Table F-3.

The Atomic Energy Commission (AEC) was split in January 1975 to form the ERDA and the Nuclear Regulatory Commission (NRC). The radiation protection standards of the AEC Manual Chapter 0524, dated November 8, 1968, are shown in Table F-4 (AEC, 1968). The November 8, 1968, version superseded the earlier version dated August 12, 1963 (AEC, 1963), which contained essentially the same radiation protection standards. Thus, the standards in Table F-4 were in effect from 1963 to 1976.

On December 19, 1961, the Idaho Operation Office of the AEC published Announcement No. 163, "Radiation Protection Guide for the NRTS" (AEC, 1961). The radiation protection guide values for population groups, Table F-5, are those recommended in the second report of the Federal Radiation Council (FRC) (FRC, 1961). The Radiation Protection Guide (RPG) for the whole body had been included in FRC (1960) and was 0.5 rem to the whole body for individuals and 0.17 rem to the whole body for average population groups. Thus, from 1960 through 1984, the radiation protection standards for the public were essentially the same.

The introduction to FRC (1960) is informative concerning the standards before 1960. It is quoted at length below.

> 1.1 It was recognized soon after discovery of x-rays that exposure to large amounts of ionizing radiation can produce deleterious effects on the human body so exposed. More recently, because of increased scientific knowledge and widespread use of radiation, additional attention has been directed to the possible effects of lower levels of radiation on future generations. Various scientific bodies have made recommendations to limit the irradiation of the human body. Probably the oldest of such scientific bodies are the International Commission on Radiological Protection (ICRP) and the U.S. National Committee on Radiation Protection and Measurements (NCRP). Initially, these bodies were interested primarily in the irradiation of those exposed occupationally, but recently they have been concerned with those who are nonoccupationally exposed.

Table F-4. Radiation protection standards for external and internal exposure for individuals and population groups in uncontrolled areas, 1963-1976[a]

Type of Exposure	Annual Dose or Dose Commitment: Based on Dose to Critical Individuals at Points of Maximum Probable Exposure (rem)	Annual Dose or Dose Commitment: Based on Average Dose to a Suitable Sample of the Exposed Population[b] (rem)
Whole body, gonads, or bone marrow	0.5	0.17
Other organs[c]	1.5	0.5

a. AEC (1963); AEC (1968)

b. See Paragraph, FRC (1960), for discussion on concept of suitable sample of exposed population.

c. An acceptable alternate standard for bone for individuals is the ICRP standard of 0.003 μg of radium-226 or its biological equivalent. The alternative standard for populations would be one-third this ICRP standard.

Table F-5. Guide values for population groups (AEC/ID), 1961-1962[a]

Summary of Guide Values for Population Groups

Organ	RPG for Individuals	RPG for Average of Suitable Sample Exposed Population
Thyroid	1.5 rem/yr	0.5 rem/yr
Bone Marrow	0.5 rem/yr	0.17 rem/yr
Bone	1.5 rem/yr	0.5 rem/yr
Bone (alternate guide)	0.003 microgram of radium 226 in adult skeleton or biological equivalent of radium 226	0.001 microgram of radium 226 in adult skeleton or biological equivalent of radium 226

a. AEC (1961)

FRC Report No. 1 Recommendations

	Individuals in the General Population	Average Population Groups
Whole Body	0.5 rem/yr	0.17 rem/yr

1.2 The ICRP was formed in 1928 under the auspices of the International Congress of Radiology. It is now a Commission of the International Society of Radiology. This Commission has published recommendations about every three years except for the period 1938-49.

1.3 The NCRP was initially organized as the "Advisory Committee on X-ray and Radium Protection." The initial membership included representatives from the medical societies, x-ray equipment manufacturers, and the National Bureau of Standards. After the reorganization in 1946, the name was changed to the National Committee on Radiation Protection and Measurements, and additional representatives from other organizations having scientific interest in the field were included. The recommendations of this group have generally been published as National Bureau of Standards handbooks. Since 1947, 15 such handbooks have been made available on different aspects of the protection problem.

1.4 In 1956, the National Academy of Sciences-National Research Council published reports of its Committees on the Biological Effects of Atomic Radiation. For genetic protection this group recommended a maximum gonadal dose up to age 30 both for individual radiation workers and for the entire population. These committees published a revised report in 1960.

1.5 The recommendations of the NCRP, ICRP, and NAS-NRC are in rather close agreement. The recommendations of the NCRP have received wide acceptance in the United States.

1.6 In 1955, The United Nations established a Scientific Committee on The Effects of Atomic Radiation (UNSCEAR). The report of this group (UNSCEAR, 1958) summarized the current knowledge on effects of radiation exposure and on human exposure levels. The report also contained predictions on exposures from testing of nuclear devices under various assumptions.

1.7 The Joint Committee on Atomic Energy of the Congress held public hearings in 1957 on "The Nature of Radioactive Fallout and Its Effects on Man." The same committee held hearings in 1959 on "Industrial Radioactive Waste Disposal;" on "Employee Radiation Hazards and Workman's Compensation;" on "Fallout from Nuclear Weapons Tests;" and on "Biological and Environmental Effects of Nuclear War." In all these hearings, questions of the biological effects of radiation and of protection against excessive exposure to radiation received attention.

1.8 The Federal Radiation Council was formed in 1959 (Public Law 86-373) to provide a Federal policy on human radiation exposure. A major function of the Council is to ". . . advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with States . . . ." This staff report is a first step in carrying out this

> responsibility. As knowledge of the biological effects of radiation increases, and as factors making exposure to radiation desirable undergo change, modifications and amplifications of the recommendations of this staff report probably will be required.

FRC (1960) refers to the wide acceptance of NCRP reports. According to a letter from Lauriston S. Taylor, dated September 15, 1949, the AEC had looked to the NCRP for recommendations on radiation protection and had not attempted to establish its own radiation safety standards (Taylor, 1979).

On April 15, 1958, the NCRP published an addendum, entitled "Maximum Permissible Radiation Exposures to Man," to the National Bureau of Standards (NBS) Handbook 59 (DOC, 1958). (This addendum replaced a preliminary statement of the NCRP issued on January 8, 1957). At that time, the concept of limiting the radiation exposure to a suitable sample of the population had not been developed, and the recommendations addressed the dose to an individual outside of the controlled area. Specifically, the addendum states,

> The radiation or radioactive material outside a controlled area, attributable to normal operations within the controlled area, shall be such that it is improbable that any individual will receive a dose of more than 0.5 rem to any 1 year from external radiation.
>
> The maximum permissible average body burden of radionuclides in persons outside of the controlled area and attributable to the operations within the controlled area shall not exceed one-tenth of that for radiation workers.[7] This will normally entail control of the average concentrations in air or water at the point of intake, or rate of intake to the body in foodstuffs, to levels not exceeding one-tenth of the maximum permissible concentrations allowed in air, water, and foodstuffs for occupational exposure. The body burden and concentrations of radionuclides may be averaged over periods up to 1 year.
>
> The maximum permissible dose and the maximum permissible concentrations of radionuclides as recommended above are primarily for the purpose of keeping the average dose to the whole population as low as reasonably possible, and not because of specific injury to the individual.

Note the recommendation that concentrations in air or water not exceed one-tenth of the maximum permissible concentration for occupational exposures. Dose standards for specific organs may be derived by using one-tenth of the

occupational standard. The recommended dose standards for an individual can be summarized as 0.5 rem to the whole body; 3 rem to the thyroid, skin, and bone; and 1.5 rem to other organs and were applied from 1958 through 1960.

Before the publication of Announcement No. 163 by the Idaho Operations Office (AEC, 1961) and the April 15, 1958, NCRP Addendum to NBS Handbook 59, the radiation protection standards of AEC Manual Chapter 0524, dated February 1, 1958 (AEC, 1958) were in effect. This manual chapter adopted the recommendations contained in the preliminary statement of the NCRP made on January 8, 1957, titled "Maximum Permissible Radiation Exposure to Man" (NBS, 1957). Although the preliminary statement did not address the thyroid, skin, and other organs of the body, it included the same dose standard for the whole body as the April 15, 1958, NCRP Addendum to NBS Handbook 59. Thus, for 1957, the dose standard for an individual would have been 0.5 rem to the whole body, 2.5 rem to the skin, and 1.5 rem to other organs.

NBS Handbook 59, *Permissible Dose from External Sources of Ionizing Radiation*, was issued on September 24, 1954 (DOC, 1954). As stated in the Handbook's preface, the formal publication was "delayed for a considerable time; even though the substance of the committee's recommendations has been well known and in use for a number of years." (A preliminary report by this NCRP committee was prepared during the summer of 1949; except for details, it did not differ greatly from the report published in 1954.) Although the concept of a dose standard for the general population had not been developed, the NCRP report recommended that the dose to minors exposed to radiation in the course of their normal activities should not exceed one-tenth of the respective permissible weekly doses specified for occupational workers. Numerically, this equated to 1.5 rem in 1 year to the whole body of minors (including the blood-forming organs and lens of the eye). By generalizing the one-tenth limitation to internal exposures, the permissible internal dose to organs other than the skin would correspond to 1.5 rem (DOC, 1953). The permissible skin dose would correspond to 2.5 rem. Based on the above discussion, these standards would be applicable from 1952 to 1956.

In summary, the whole body (including blood-forming organs and the gonads) radiation protection standards for the maximally exposed individual

applicable to DOE operations at the INEL can be summarized as 25 mrem/yr from 1985 to 1987 from airborne emissions only, 500 mrem/yr from 1957 to 1984, and 1500 mrem/yr from 1952 to 1956. Similarly, the standards for exposure of the critical organ were 75 mrem/yr from 1985 to 1987 from airborne emissions only, and 1500 mrem/yr from 1961 to 1984. From 1958 to 1960, the standards for exposure to the critical organ were 3000 mrem/yr to the thyroid, skin, and bone and 1500 mrem/yr to other organs. From 1952 to 1957, the standards for exposure to the critical organ can be inferred to be 2500 mrem/yr for the skin and 1500 mrem/yr for other organs.

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APPENDIX G
QUALITY ASSURANCE

CONTENTS

G.1 QUALITY ASSURANCE G-5

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G.1 QUALITY ASSURANCE

During the preparation of the *Draft Idaho National Engineering Laboratory Historical Dose Evaluation*, quality assurance (QA) activities were performed to help ensure that the information in the report and dose calculations were accurate. However, the independent Peer Review Panel noted that the QA activities had not been formally documented. The Panel also indicated that QA activities associated with data verification and internal review needed to be increased.

To address the recommendations of the Peer Review Panel and improve the quality of the final report, a QA plan was developed at the beginning of the revision of the draft report. The activities necessary to perform the INEL Historical Dose Evaluation were divided into the seven elements shown in Figure G-1. These elements involved activities necessary to establish the (1) source term, (2) dispersion factors, (3) dose conversion factors, (4) ingestion uptake factors, (5) radionuclide half-lives, (6) deposition velocities, and (7) exposure scenario parameters. These elements were further divided into individual tasks identified as requiring (a) documentation of references and/or primary sources of data, (b) transcription of data or other information, (c) consideration of the appropriateness of the data and assumptions used to evaluate the data, and (d) verification of calculations (see Figures G-2 through G-10). Individuals were assigned responsibilities for each of the major elements in the preparation of the final INEL Historical Dose Evaluation.

QA files were prepared to document the performance of identified tasks and provide a record of pertinent references in the literature; basic data considered; assumptions, calculations, and models used to evaluate data; results of the evaluation; and tabulations of important data. QA files often provided documentation of several individual tasks. QA files became important reference materials to members of the INEL Historical Dose Evaluation Task Group in the preparation of the final report, helping to ensure the consistent, accurate descriptions of activities performed. Finally, the QA files provide an important, permanent record that can be consulted for future verification or dose reconstruction efforts.

A cover sheet was attached to each QA file (Figure G-11) indicating the major element, the specific task performed, and the person preparing the file. Each QA file was then reviewed by an individual not responsible for performing the task. The review involved either a spot check or a complete (100%) review of the information provided in the QA file, and the determination that the information was adequate to support the performance of the task.

The Task Group met weekly to discuss problems, report on the status of individual tasks, and to make decisions on important approaches and methods. When the dose calculations were completed, each member of the Task Group carefully reviewed the text of the final report, checking for completeness and accuracy.

Copies of the QA files are in the INEL Public Reading Room. In addition, the INEL Public Reading Room and the INEL Technical Library contain copies of the documents referenced in this report so that they are readily available for use by interested individuals.

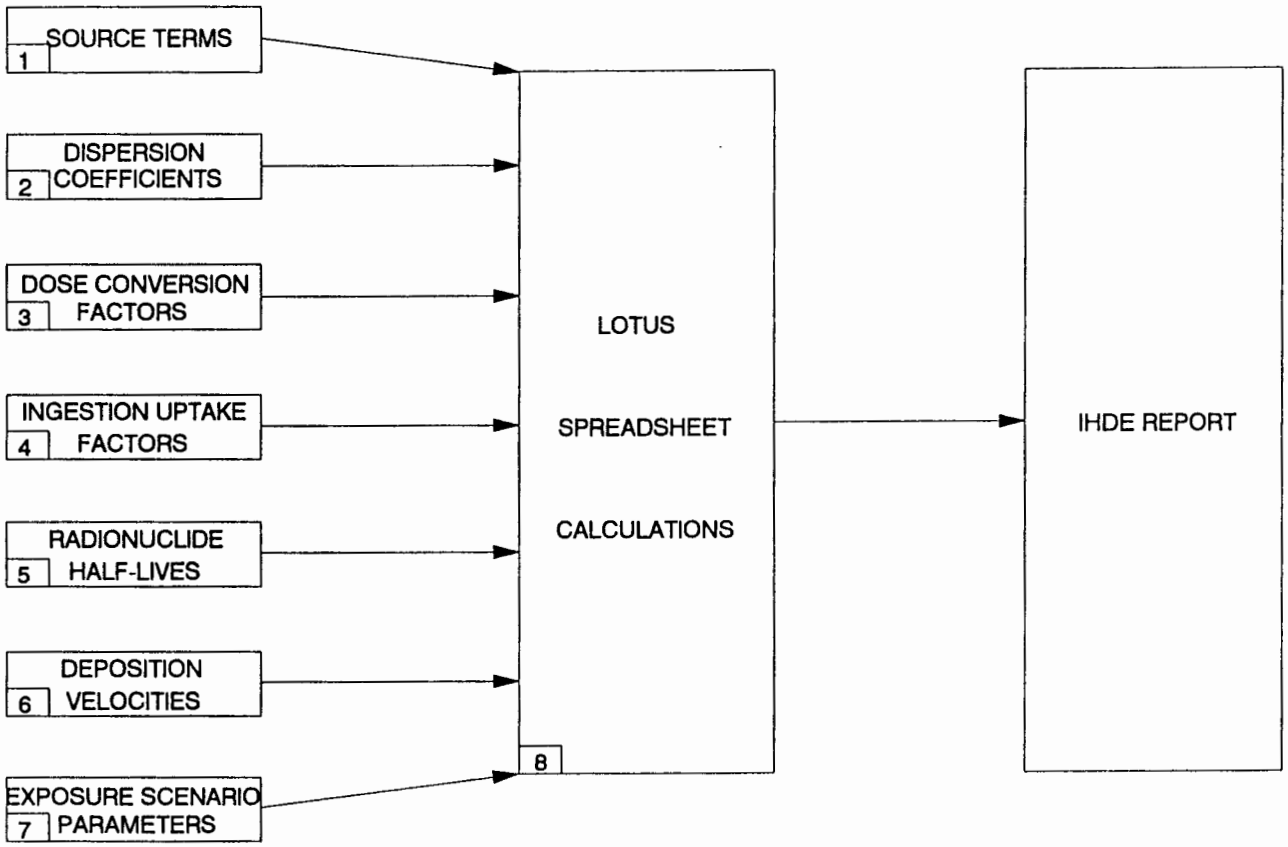

Figure G-1. Elements of INEL historical dose calculations.

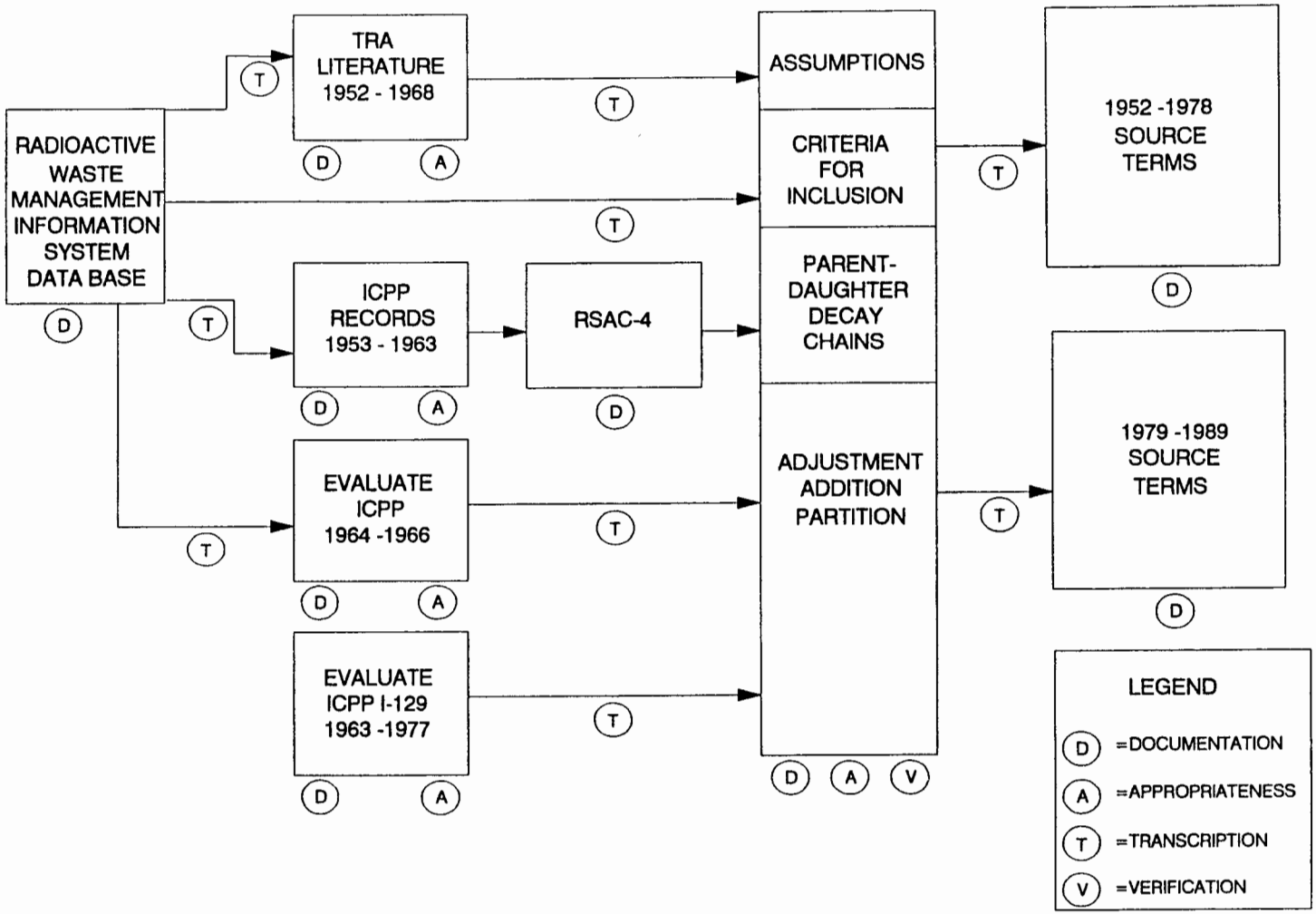

Figure G-2. Source term quality control tasks (operational releases).

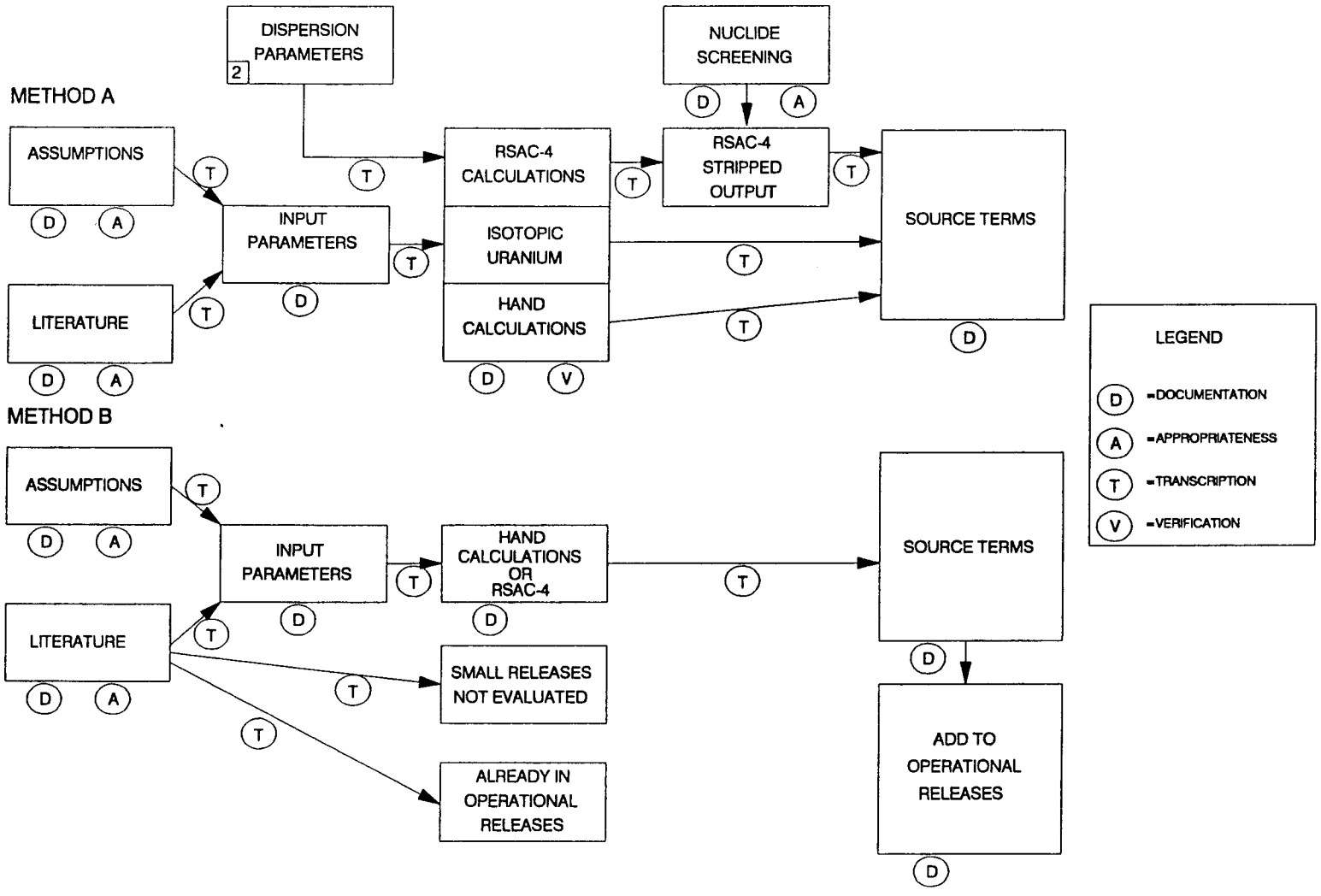

Figure G-3. Source term quality control tasks (episodic releases).

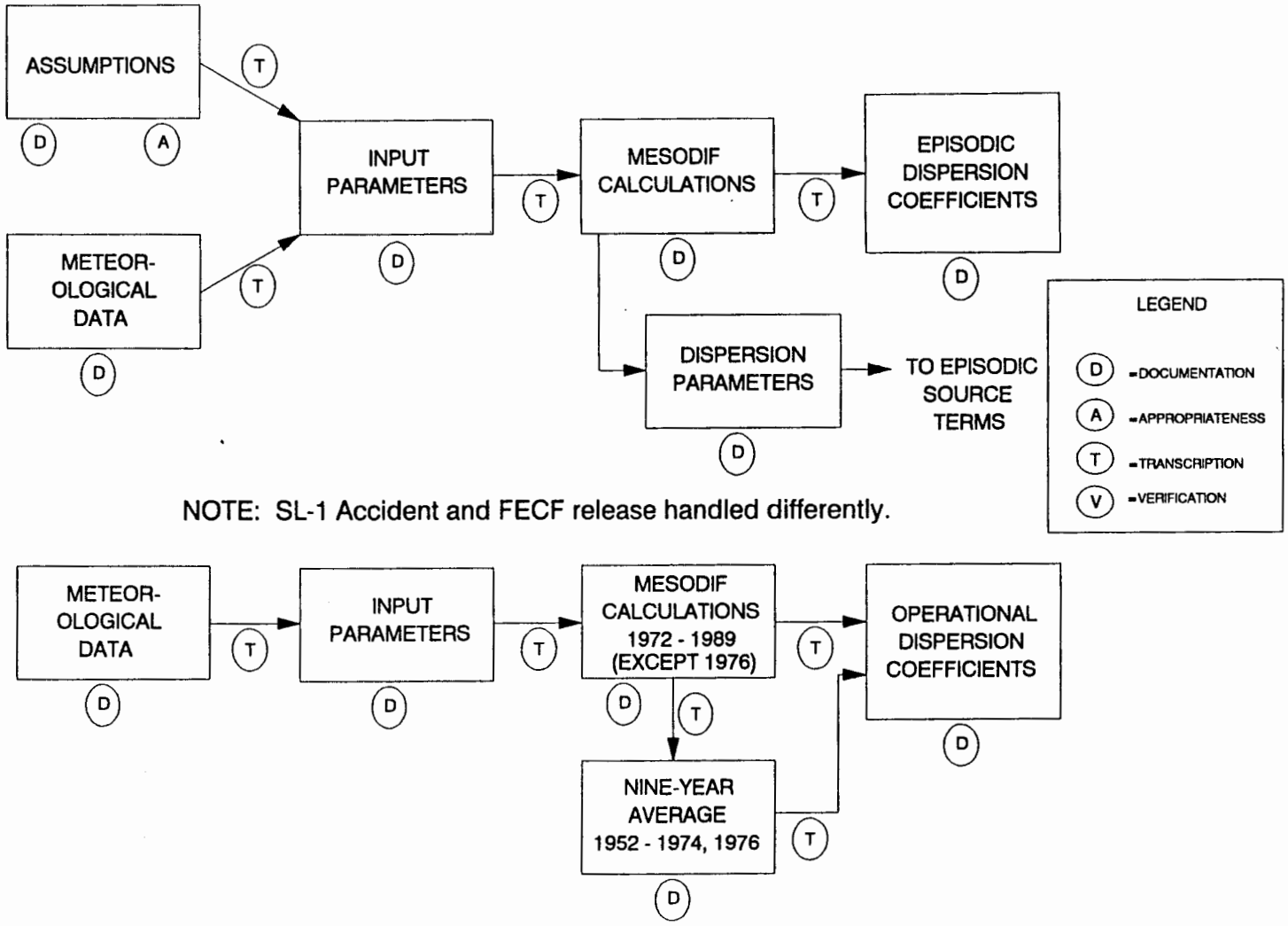

Figure G-4. Dispersion coefficient quality control tasks.

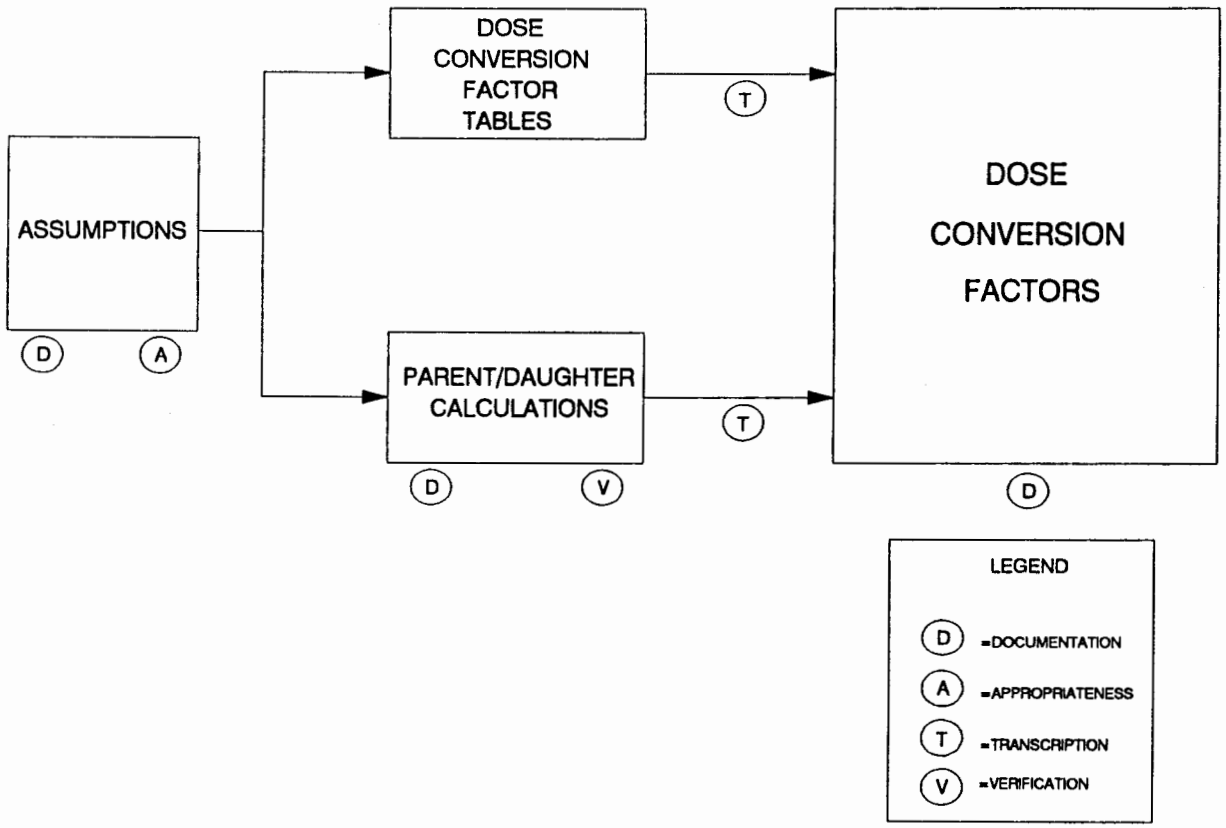

Figure G-5. Dose conversion factors quality control tasks.

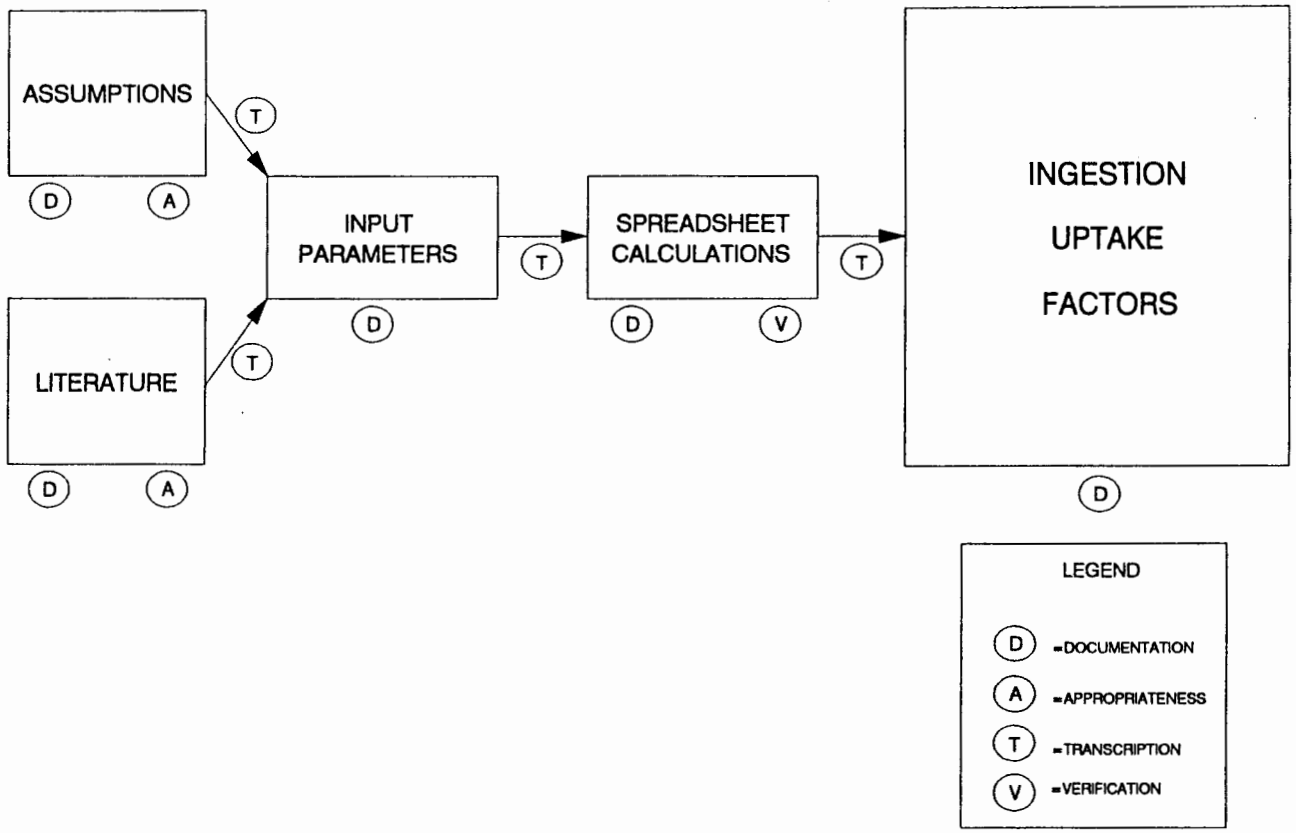

Figure G-6. Ingestion uptake factors quality control tasks.

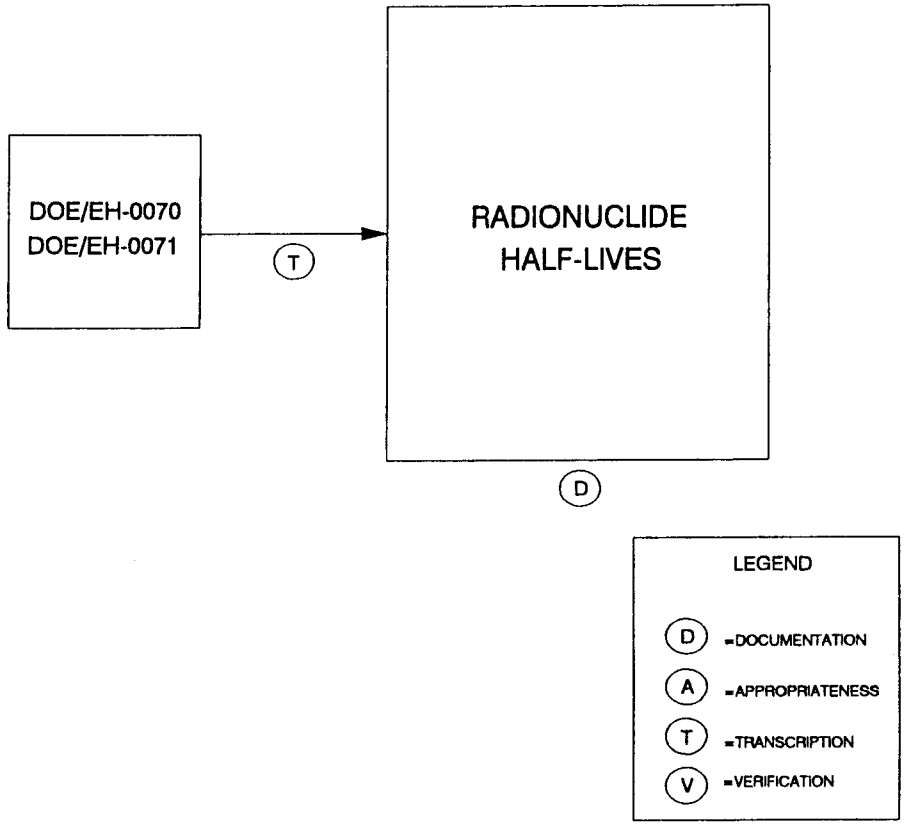

Figure G-7. Radionuclide half-lives quality control tasks.

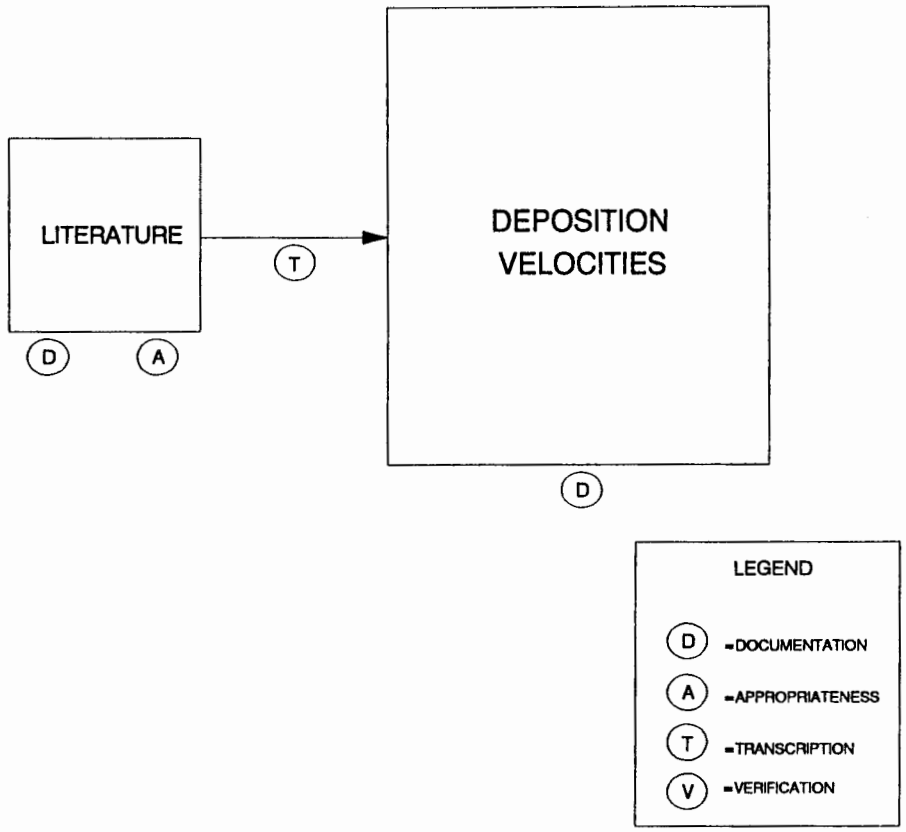

Figure G-8. Deposition velocities quality control tasks.

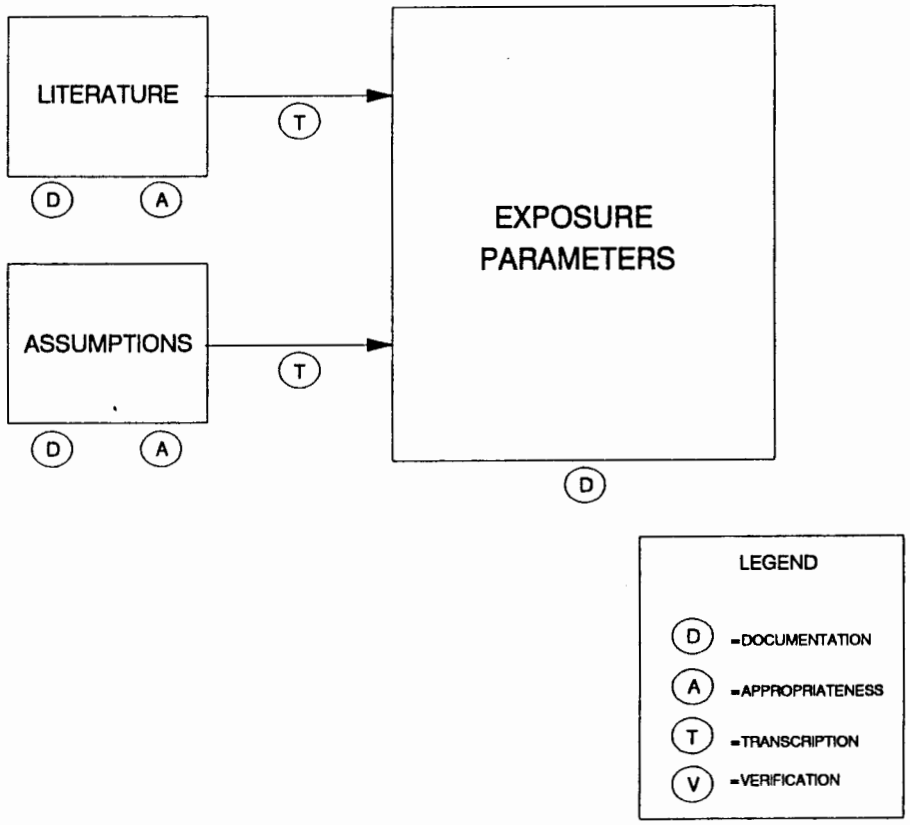

Figure G-9. Exposure scenario parameters quality control tasks.

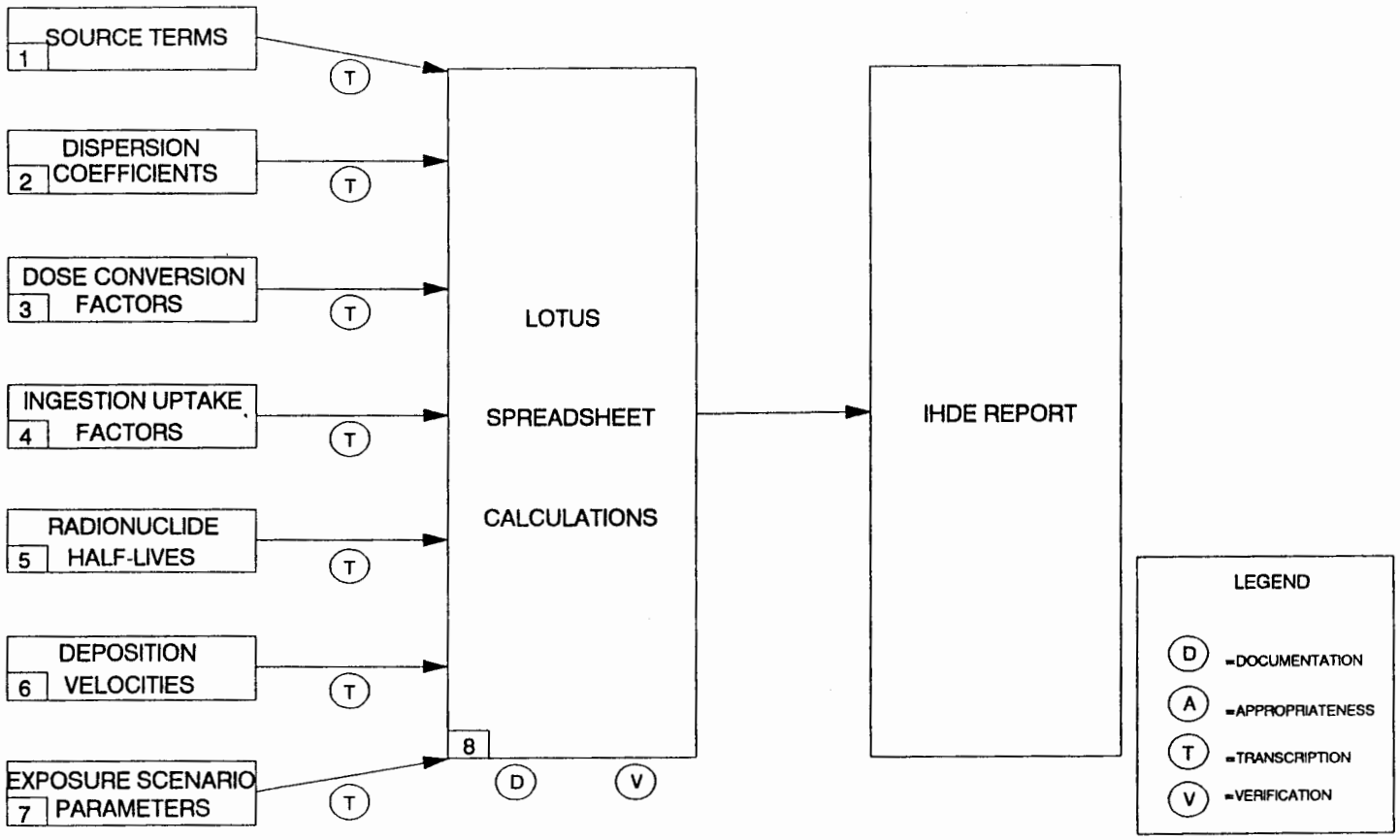

Figure G-10. Lotus spreadsheet and report quality control tasks.

INEL Historical Dose Evaluation
Quality Assurance File

- ☐ Source Term Quality Control Tasks (Operational Releases)
- ☐ Source Term Quality Control Tasks (Episodic Releases)
- ☐ Dispersion Coefficient Quality Control Tasks (Operational)
- ☐ Dispersion Coefficient Quality Control Tasks (Epidsodic)
- ☐ Dose Conversion Factors Quality Control Tasks
- ☐ Ingestion Uptake Factors Quality Control Tasks
- ☐ Radionuclide Half-lives Quality Control Tasks
- ☐ Deposition Velocities Quality Control Tasks
- ☐ Exposure Scenario Parameters Quality Control Tasks
- ☐ Lotus Spreadsheet and Report Quality Control Tasks

Operation

- ☐ Documentation
- ☐ Appropriateness
- ☐ Transcription
- ☐ Verification

Prepared by _____ Date _____

Quality Assurance Review

- ☐ 100%
- ☐ Spot: _____%

Reviewed by _____ Date _____

Page ____ of ____

Figure G-11. Quality assurance file cover sheet.

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Report Date: 18-May-91
Infant receptor

9/17/58 Episodic Release
FPFRT-7

Dose Equivalent From Ingestion (mrem)

Radionuclide	EDE	Skin	Thyroid	Lungs	Liver	Bone Surface	Red Marrow
Sr-89	6.64E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	6.19E-02	5.24E-03	5.24E-03	5.24E-03	5.24E-03	4.76E-01	3.38E-01
Y-91	1.24E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.13E-02	7.48E-04	4.32E-04	7.48E-04	1.25E-03	2.61E-02	7.14E-03
Ru-103	2.86E-04	3.65E-05	4.09E-05	4.09E-05	5.57E-05	5.20E-05	5.95E-05
Ru-106	3.76E-04	9.30E-05	9.73E-05	9.73E-05	9.73E-05	1.14E-04	1.52E-04
Te-132	6.38E-09	0.00E+00	1.90E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131 (ELE)	4.31E-01	1.84E-03	1.45E+01	3.49E-03	1.80E-03	2.59E-03	2.74E-03
I-132	6.00E-13	5.00E-14	1.37E-11	1.00E-13	8.00E-14	7.00E-14	8.00E-14
Cs-137	1.93E-02	1.54E-02	1.83E-02	1.73E-02	1.83E-02	2.02E-02	2.60E-02
Ba-140	2.49E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	4.03E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	4.16E-03	8.32E-05	8.32E-05	8.32E-05	1.25E-03	3.58E-03	4.58E-03
Pr-143	7.39E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	2.86E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTALS	5.37E-01	2.35E-02	1.45E+01	2.70E-02	2.80E-02	5.29E-01	3.79E-01