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1983 ENVIRONMENTAL MONITORING PROGRAM REPORT FOR IDAHO NATIONAL ENGINEERING LABORATORY SITE

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PREFACE

This report presents the offsite data collected in 1983 for the routine environmental monitoring program conducted by the Department of Energy's Radiological and Environmental Sciences Laboratory (RESL/ID) at the Idaho National Engineering Laboratory (INEL) Site. In some cases onsite data may be reported if needed to clarify or interpret offsite monitoring results or to demonstrate compliance with environmental standards and regulations. The purpose of this routine program is to monitor radioactive and nonradioactive materials resulting from INEL Site operations which may reach the surrounding offsite environment and population. This report is prepared in accordance with the Department of Energy requirements in DOE Order 5484.1 and is not intended to cover the numerous special environmental research programs being conducted at the INEL by the RESL/ID and others. Generally, these latter programs are aimed at quantifying the effects of Site operations on the onsite environment.

Note: Use of commercial product names is for accuracy in technical reporting and does not constitute endorsement of the product by the United States Government.

SUMMARY

The results of the various monitoring programs for 1983 indicated that radioactivity from the Idaho National Engineering Laboratory (INEL) Site operations could not be distinguished from worldwide fallout and natural radioactivity in the region surrounding the Site. Although some radioactive materials were discharged during Site operations, concentrations and doses to the surrounding population were of no health consequence and were far less than State of Idaho and Federal health protection guidelines. This report describes the air, water, and foodstuff samples routinely collected at the INEL boundary locations and at locations distant from the INEL Site. The report also compares and evaluates the sample results and discusses implications.

There was no statistical difference in particulate beta concentrations in air as measured at Site boundary stations and those measured at distant sampling stations. Air samples were also analyzed for specific radionuclides. Some radionuclides were detected, but their presence was attributable to natural sources, to worldwide fallout or to statistical variations and not to Site operations. Monitoring for SO_2 and NO_2 at an onsite location during the year showed that concentrations of these nonradiological pollutants were far below U. S. Environmental Protection Agency (EPA) standards.

A few samples of offsite well water and surface water contained concentrations of gross alpha or gross beta activity near the minimum detectable concentration levels, but these were probably due to natural radioactivity. All offsite water samples were considerably below the EPA maximum contaminant level for community drinking water systems. No offsite water samples contained detectable tritium concentrations.

A few milk samples from scattered areas contained I-131 concentrations near the minimum detectable concentration level, but these were not attributable to Site operations. Some milk and lettuce samples, and all wheat samples, contained small amounts of Sr-90. A few wheat and lettuce samples also had detectable concentrations of Cs-137. The presence of both nuclides is probably due to the deposition of these nuclides on soil as a result of worldwide fallout.

Ionizing radiation measured simultaneously at Site boundary and distant locations showed only natural background levels.

For details on monitoring results, see the section "Monitoring Data Collection, Analyses and Evaluation."

Measurable amounts of radioactivity, primarily in the form of noble gases, are released from various plant facilities and subsequently travel offsite. When they reach the Site boundary, these gases are in such low concentrations that their effect on radiation levels cannot be measured; but their contributions to offsite 50-year dose commitments are nevertheless calculated.

A hypothetical maximum 50-year whole-body dose commitment on the southern Site boundary was calculated to be 0.02 millirem (mrem). The calculation assumed continuous submersion in and inhalation of radioactivity in air, and exposure to radioactive particulates deposited on soil at that location. This hypothetical dose commitment is about 0.014% of the natural background radiation dose rate of about 140 mrem per year in this area. The potential maximum dose commitment to an individual living nearest the Site was calculated to be 0.014 mrem at Atomic City, Idaho. This also represents the maximum potential dose commitment to a member of a population group. The maximum potential population dose commitment from submersion, inhalation, and deposition to the approximately 114,600 people residing within an 80-km (50-mi) radius from the center of the TRA-ICPP area of the INEL Site was estimated to be 0.10 man-rem. This dose commitment is about 0.0006% of the population dose from natural background radioactivity, which is estimated to be 16,000 man-rem. These 50-year dose commitments and their implications are discussed in the section "Radiological Impact of INEL Site Operations."

Calculations indicate that the maximum potential 50-year dose commitment to an individual from indirect exposure pathways due to ingestion of wild game animals is about 2% of the radiation protection standard for individuals at points of maximum probable exposure. See the section "Environmental Standards and Regulations." The potential man-rem dose commitment to all offsite populations from these exposure pathways would realistically be less than the dose commitment from submersion, inhalation, and deposition pathways due to the very small probability that an individual in the population would consume an animal containing significant amounts of radioactivity.

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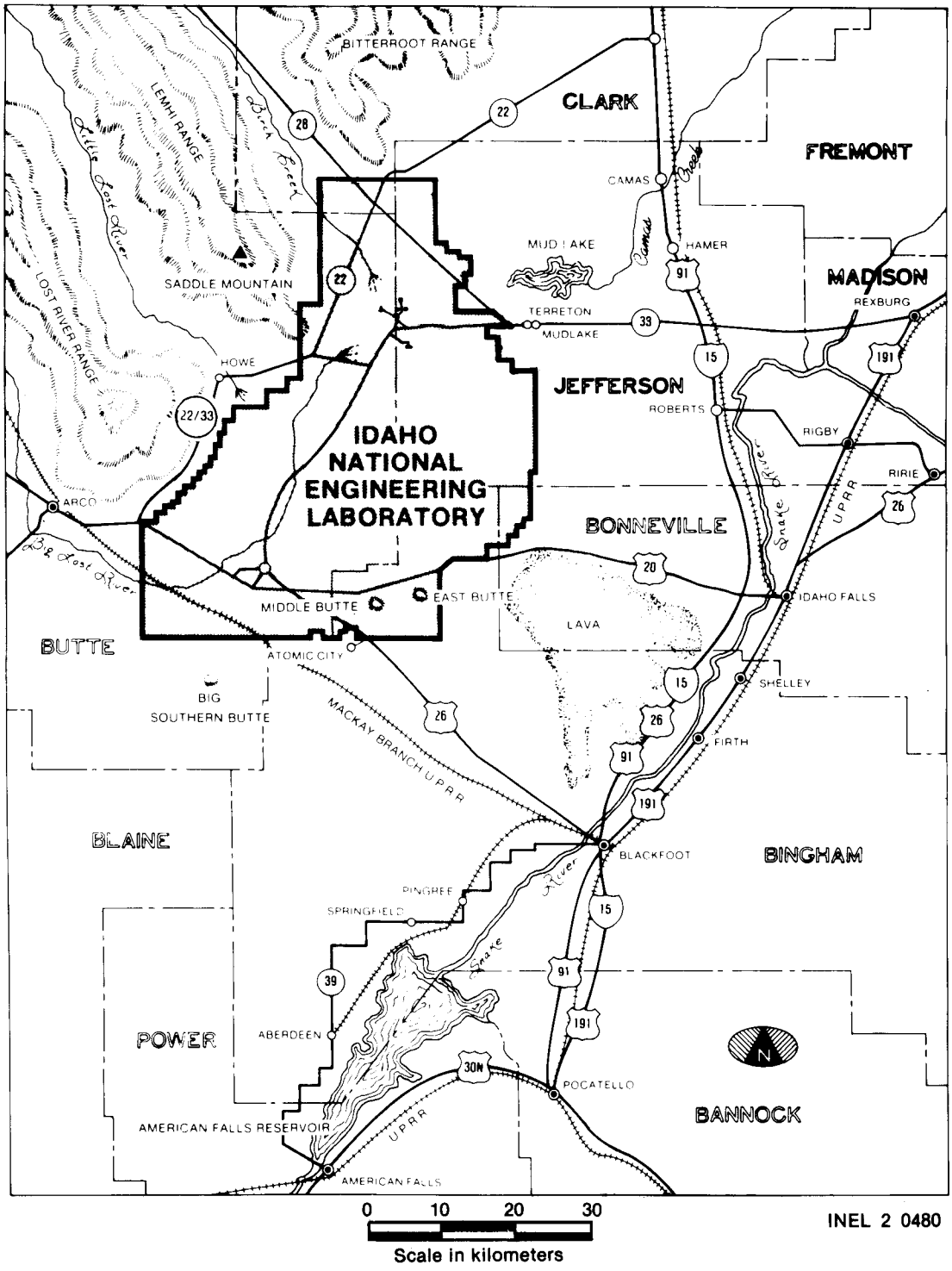

Figure 1. INEL Site vicinity map.

1983 ENVIRONMENTAL MONITORING PROGRAM REPORT FOR IDAHO NATIONAL ENGINEERING LABORATORY SITE

INTRODUCTION

The Department of Energy's (DOE) Idaho National Engineering Laboratory (INEL) was established by the Federal Government in 1949 to conduct research and further the development of nuclear reactors and ancillary equipment. The 2300-km^2 (890-mi^2) Site is located west of Idaho Falls, Idaho on a high desert plain (see Figures 1 and 2). In 1975 the Site was also designated as one of the nation's five National Environmental Research Parks (NERP). A more detailed description of the Site location, environment, and current major activities is given in Appendix A.

Figure 2. North-facing view of part of the INEL Site.

MONITORING DATA COLLECTION, ANALYSES AND EVALUATION

General

During normal operation of the reactors and the fuel reprocessing plant at the Site, some radioactivity is released to the environment. The environmental pathways by which radioactive materials may be transported from the Site to nearby populations are directly through atmospheric transport or indirectly through soils, foodstuffs, or animals. No evidence is available that radionuclides in the water of the Snake River Plain aquifer have migrated beyond the INEL southern boundary.

The environmental monitoring program for the Site and vicinity for 1983 included the collection and analysis of samples from potential exposure pathways. Table I gives a summary of the offsite program. Measurements at Site boundary locations are compared to measurements at distant locations to assess the impact of INEL Site operations on the offsite environment. Concentrations of radioactive and some nonradioactive pollutants in the environment are compared to applicable standards and guides and to background and natural radioactivity. Most radioactive concentrations in this report are compared to the concentration guides for uncontrolled areas listed in "Requirements for Radiation Protection," *DOE Order 5480.1A*, Chapter XI. See the section "Environmental Standards and Regulations."

Air and water were routinely monitored for radioactivity at a number of onsite, as well as boundary and distant locations. Concentrations of radionuclides in milk, wheat, and lettuce samples were measured at Site boundary and distant locations. Offsite soils are only sampled in even-numbered years. Environmental radiation exposure rates (cumulative from November 1982 to November 1983) were measured at Site boundary and distant locations. See Appendix B for a description of the quality control and assurance program maintained by the Radiological and Environmental Sciences Laboratory.

A discussion of each routine program follows. For each program a presentation and interpretation of the data are given, as are the location of each sampling station and the number of samples collected. See Appendix C for a discussion of the statistics used to analyze the data in this report.

Air Sampling

Radiological. Airborne particulate radioactivity is monitored offsite by a network of 10 continuous air samplers at locations shown in Figure 3. Air samplers are located in the small communities close to the Site boundary and at the more distant locations of Blackfoot, the Craters of the Moon National Monument, and Idaho Falls, Idaho. These distant or background locations are in directions usually crosswind to the Site and are sufficiently remote to ensure that radioactivity detected is due to natural background or sources other than Site operations. The whole network provides comprehensive surveillance of atmospheric radioactivity and theoretically makes it possible to differentiate Site releases from worldwide fallout and long-lived natural radioactivity.

Each air sampler (see Figure 4) maintains an average air flow of about 40 L/min (1.5 ft^3/min) through a set of filters consisting of a membrane filter (Gelman Model VP-1200) followed by an activated charcoal-impregnated cellulose fiber filter (Gelman Model AC-1). The filters are 99% efficient for airborne particulate radioactivity and elemental iodine vapor. One offsite and two onsite locations also have samplers for tritium in water vapor. In these samplers air is passed through a column of silica gel at a rate of 0.3 L/min (0.65 ft^3/hr). Noble gases (argon, krypton, and xenon) are monitored at their onsite release points only.

The filters are collected weekly and analyzed after waiting a minimum of five days to allow the naturally occurring short-lived radon and thoron daughters to decay. Gross beta analysis is performed on each filter in a low background beta counter. If the beta activity on a membrane filter exceeds about 1×10^{-12} μCi/mL, the filter is analyzed by gamma spectrometry. All activity detected on the charcoal-impregnated filters is initially assumed to be I-131. If the beta activity on the charcoal filter exceeds about 7×10^{-14} μCi/mL, the filter is analyzed by gamma spectrometry to determine the I-131 component. At the end of each quarter, the membrane filters are composited according to location. The composited samples from each location are analyzed for specific radionuclides by gamma spectrometry. One composite from a distant location is analyzed each quarter for

TABLE I
OFFSITE MONITORING PROGRAM SUMMARY

Medium Sampled	Type of Analysis	Frequency of Analysis	Sample Size	Count Time (min)	∿Minimum Detectable Concentration (MDC)
Air	Gross beta	Weekly	1 to 4 x 10^8 mL	20	8 x 10^{-15} μCi/mL
	HTO[a]	3 to 7 weeks	1 to 10 x 10^6 mL	100	1 x 10^{-11} μCi/mL
	Specific gamma	Quarterly	3 to 5 x 10^9 mL	60	1 to 10 x 10^{-15} μCi/mL
	Pu	Quarterly	3 to 5 x 10^9 mL	1000	6 x 10^{-18} μCi/mL
	Am	Quarterly	3 to 5 x 10^9 mL	1000	8 x 10^{-18} μCi/mL
	Sr-90	Quarterly	3 to 5 x 10^9 mL	20	1 x 10^{-15} μCi/mL
Water	Gross alpha	Semiannually	100 mL	60	3 x 10^{-9} μCi/mL
	Gross beta	Semiannually	250 mL	20	5 x 10^{-9} μCi/mL
	HTO	Semiannually	10 mL	20	4 x 10^{-7} μCi/mL
Milk	I-131	Monthly[b]	3800 mL	1000	1 x 10^{-9} μCi/mL
	Sr-90	Annually	1000 mL	20	2 x 10^{-9} μCi/mL
	H-3	Annually	10 mL	100	4 x 10^{-7} μCi/mL
	I-129	Annually[c]	3800 mL	10	6 x 10^{-11} μCi/mL
Wheat	Specific gamma	Annually	2500 g	1000	4 x 10^{-9} μCi/g
	Sr-90	Annually	500 g	20	4 x 10^{-9} μCi/g
Lettuce	Specific gamma	Annually	30 g (dry wt)	1000	2 x 10^{-7} μCi/g
	Sr-90	Annually	30 g (dry wt)	20	8 x 10^{-8} μCi/g
Soil	Specific gamma	Biennially	400 g[d]	1000	4 x 10^{-8} μCi/g
	Pu	Biennially	10 g[d]	1000	2 x 10^{-9} μCi/g
	Am	Biennially	10 g[d]	1000	3 x 10^{-9} μCi/g
	Sr-90	Biennially	10 g[d]	100	9 x 10^{-8} μCi/g
Direct radiation exposure	Thermoluminescent dosimeter	Semiannually	5 TLDs per dosimeter	NA[e]	5 mR

a. Tritiated water.

b. One dairy is sampled weekly.

c. Three locations only.

d. Aliquot from a composited 2000-g sample.

e. NA - not applicable.

specific alpha-emitting radionuclides by chemical separation followed by alpha spectrometry. Five other composites are analyzed on a rotating schedule for specific alpha-emitting radionuclides and for Sr-90. Analyses for Sr-90 utilize chemical separation techniques followed by beta counting.

The particulate beta activity measured at the boundary locations was not distinguishable from worldwide fallout and naturally-occurring radioactivity as measured at the distant locations (see Table II). The average monthly concentrations of particulate beta activity are shown in Figure 5 for 1978 through 1983. None of the charcoal filters at any offsite location had an activity above the action level of approximately 7 x 10^{-14} μCi/mL.

The quantity and identity of radionuclides released from Site facilities is reported monthly in the Radioactive Waste Management Information System.[1] Therefore, specific radionuclide analysis is a more sensitive indicator than beta analysis of the impact of Site operations on the environment. Reported concentrations of radionuclides such as Ce-144, Ce-141, Ru-103, Ru-106, Cs-137, Sr-90, Am-241, and Pu-239/240, which may be associated with either worldwide fallout or Site operations were present at one or more locations during 1983. Comparisons of concentrations of these radionuclides to releases from Site facilities led to the conclusion that their presence was not due to Site operations. A summary of the results of specific nuclide analyses of the filter composites for distant

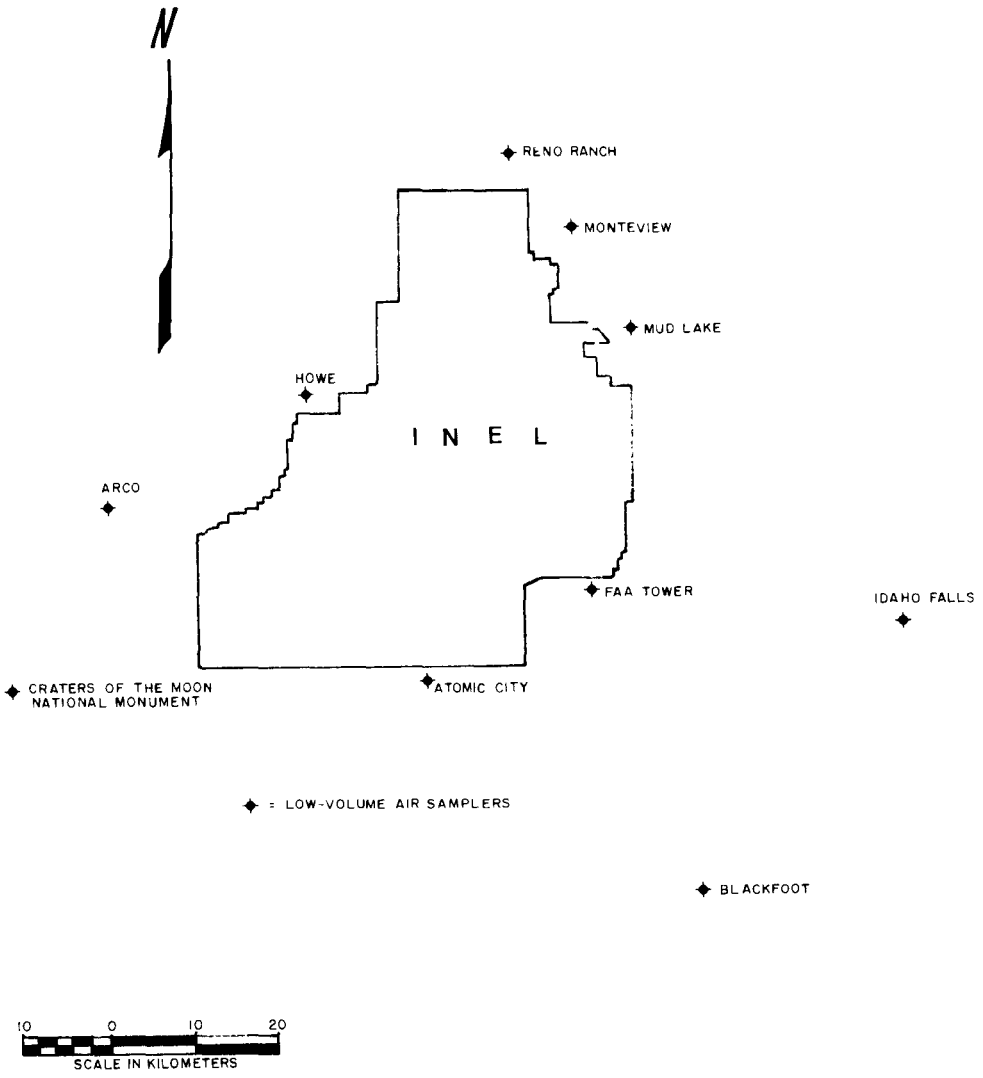

Figure 3. INEL Site air sampling network.

and boundary air sampling locations is shown in Table III. Beryllium-7, a radionuclide produced by the interaction of cosmic radiation and nitrogen in the atmosphere, is excluded.

All of the reported results of specific nuclides were very near the minimum detectable concentration. Because the concentrations are so low, it is difficult to draw firm conclusions about the source of the radioactivity (see Appendix C). Statistical tools, meteorological data and Site release information are all considered when interpreting or evaluating results that are near the minimum detectable concentration.

Two radionuclides which are not generally included in fallout were reported at concentrations near the minimum detectable concentration at one offsite location each. Antimony-125 at a concentration of $2.6 \pm 2.2 \times 10^{-15}$ μCi/mL was reported at Howe during the first quarter and Cr-51 at a concentration of $1.1 \pm 0.8 \times 10^{-14}$ μCi/mL was reported at Idaho Falls during the second quarter. Neither of the radionuclides exceeded 0.0003% of the uncontrolled area concentration guide during the quarter in which they were reported. Meteorological conditions, Site releases and onsite monitoring results were reviewed, but no relationship between Site operations and the presence of Sb-125 and Cr-51 at their respective locations could be found.

None of the nuclides detected at any offsite location had an annual boundary group mean concentration which was statistically different from the distant group mean concentration.

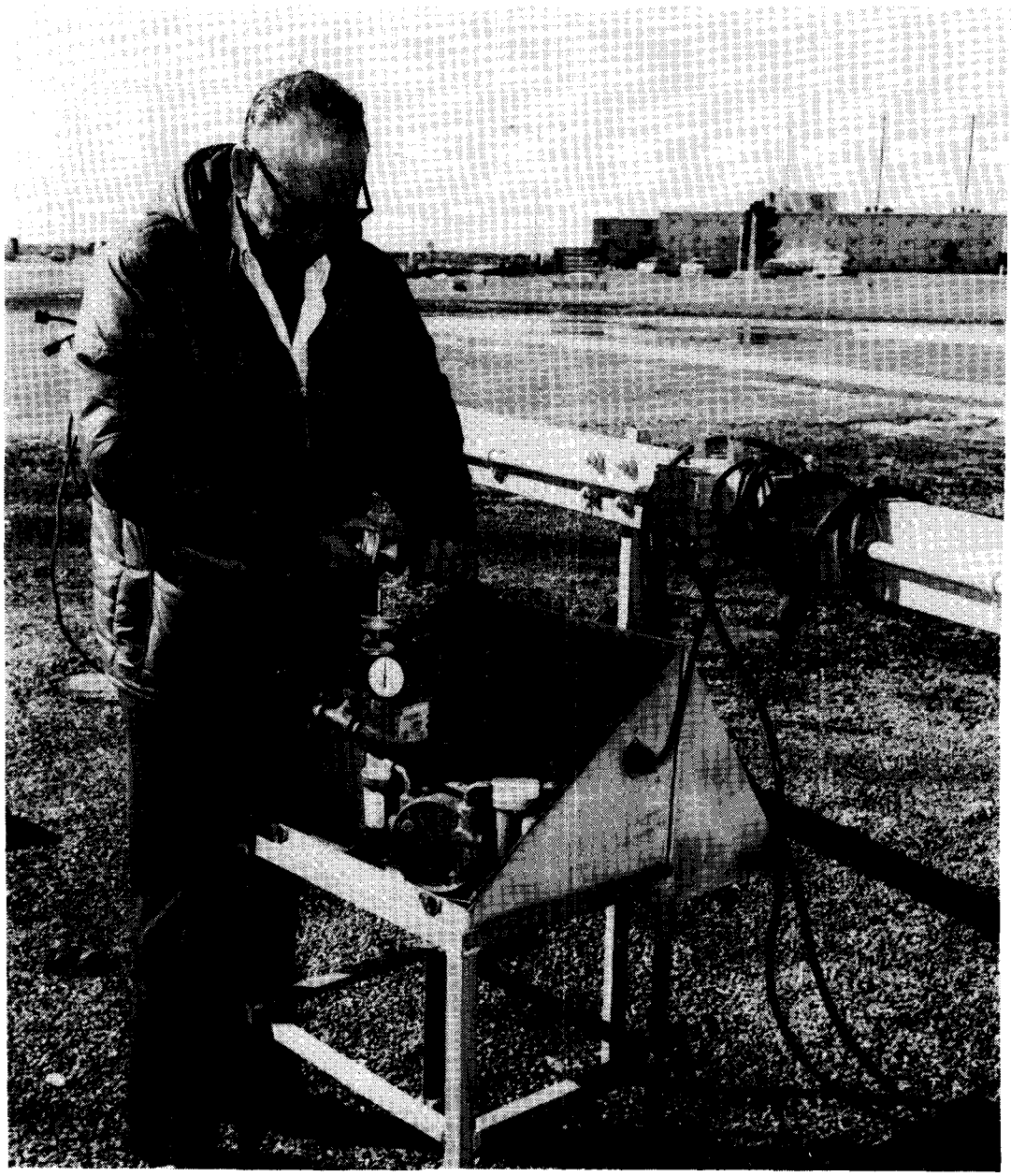

Figure 4. Low-volume air sampler used for the routine monitoring program.

Atmospheric tritium in the form of tritiated water (HTO) is monitored at Idaho Falls (a background location) and at two locations onsite. No concentration of HTO at any location exceeded the approximate minimum detectable concentration of 1×10^{-11} μCi/mL.

Nonradiological. Atmospheric particulate matter is routinely monitored at the low-volume air sampling stations using the filters previously described. The analysis involves determining the net weight of the particulate matter on the quarterly composite of weekly filters at each station. The concentrations of the samples ranged between the minimum detectable concentration and 83 $\mu g/m^3$. The boundary average was 8 $\mu g/m^3$ which was statistically the same as the distant average of 18 $\mu g/m^3$ because of the large variations between particulate matter concentrations for both groups. Most of the airborne particulates in the Site vicinity are probably windblown dust from the desert floor.

The average sulfur dioxide and nitrogen dioxide concentrations at the Site boundary were calculated

TABLE II
PARTICULATE BETA ACTIVITY IN AIR (1983)

Group	Locations	Number of Samples	Concentration (10^{-15} μCi/mL) Range	Annual Average[a]
Distant	Blackfoot	52	14-96	31 ± 4
	Craters of the Moon	51	9-83	33 ± 4
	Idaho Falls	50	12-69	29 ± 4
	Grand Mean	—	—	31 ± 2
Boundary	Arco	51	12-87	31 ± 4
	Atomic City	52	10-69	30 ± 4
	FAA Tower	52	11-69	30 ± 4
	Howe	49	13-82	33 ± 4
	Monteview	52	8-79	29 ± 4
	Mud Lake	52	14-148	35 ± 6
	Reno Ranch	52	13-73	30 ± 3
	Grand Mean	—	—	31 ± 2

a. Arithmetic mean with the 95% confidence interval for the mean. See Appendix C.

using the total 1983 discharges and a computer model of the dispersive characteristics of the air for 1983. The calculation method is the same as described in the section "Radiological Impact of INEL Site Operations—General" using mass units for releases instead of radioactivity units. A monitoring station was established during 1983 at the intersection of Van Buren Boulevard and Highway U.S. 20/26, two miles west of the Central Facilities Area (CFA) to monitor concentrations of sulfur dioxide and nitrogen oxides. At this location the average concentrations of these gases are calculated to be greater than at the southern Site boundary. The analyzers used are designated as EPA equivalent methods.

The total sulfur dioxide released in 1983 as reported in the Industrial Waste Management Information System (IWMIS) Report[2] was about 3.7×10^5 kg. Sources of sulfur dioxide are primarily Site facility heating plants and other area sources. The calculated maximum offsite concentration of sulfur dioxide near the southern Site boundary was 0.4 $\mu g/m^3$ which is well below the national primary ambient air quality standard of 80 $\mu g/m^3$. A sulfur dioxide analyzer was operated onsite at the Van Buren Boulevard location from September 12, 1983 until December 31, 1983. The analyzer operated satisfactorily during 98% of this period. The average sulfur dioxide concentration was 0.8 $\mu g/m^3$ which is 1% of the annual primary ambient air quality standard. The maximum daily average sulfur dioxide concentration during this period was 9 $\mu g/m^3$ which is 2% of the applicable primary standard. The maximum 3-hr average concentration was 26 $\mu g/m^3$ which is 2% of the applicable secondary air quality standard.

The total nitrogen oxides released in 1983 as reported in the IWMIS report[2] were about 6.0×10^5 kg. Sources of nitrogen oxides are Site operations, facility heating plants, vehicle emissions, and other area sources.

From all INEL sources the calculated maximum offsite concentration of nitrogen dioxide near the southern boundary of the Site was 0.7 $\mu g/m^3$. This concentration is well below the national primary ambient air quality standard of 100 $\mu g/m^3$. An analyzer for nitrogen oxides was operated onsite at the Van Buren Boulevard location described earlier. The analyzer was operated from April 1, 1983 until December 31, 1983. The average nitrogen dioxide concentration measured at this location was 4.2 $\mu g/m^3$, which is 4.2% of the primary ambient air quality standard. The analyzer operated satisfactorily during 79% of the period.

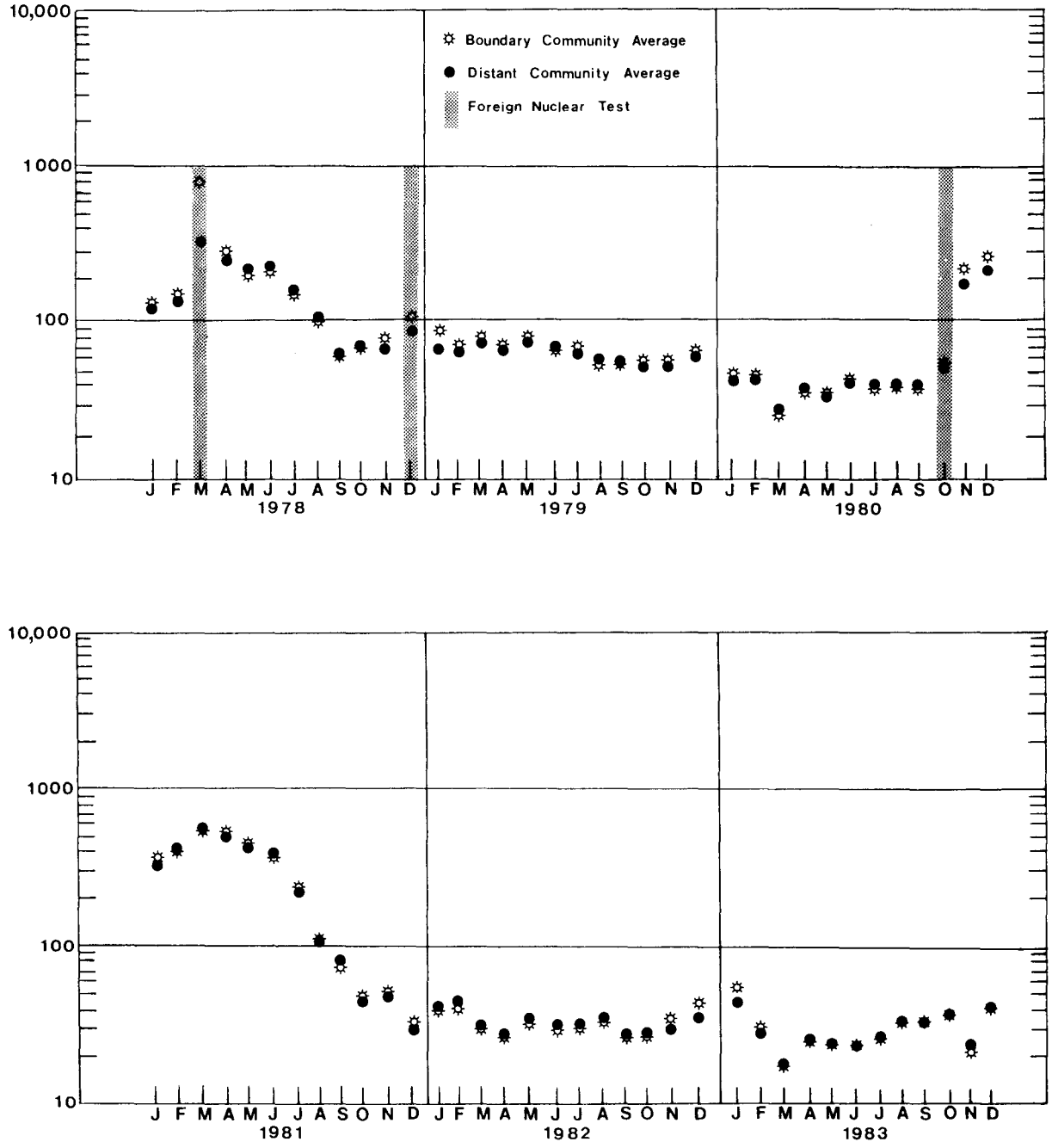

Figure 5. Particulate beta concentrations in air (10^{-15} μCi/mL).

TABLE III
SPECIFIC RADIONUCLIDE ACTIVITY IN AIR (1983)

Radionuclide	Composite Group[a]	Concentration (10^{-15} μCi/mL) Minimum[b]	Maximum[b]	Mean[c]	Concentration Guide[d]	∿MDC[e]
Ce-141	Distant	<MDC[f]	1.6 ± 1.4	NSS[g]	5,000,000	2
	Boundary	<MDC	1.7 ± 1.4	0.7 ± 0.6		
Ce-144	Distant	<MDC	<MDC	NSS	200,000	7
	Boundary	<MDC	5 ± 4	NSS		
Cs-137	Distant	<MDC	<MDC	NSS	500,000	1
	Boundary	<MDC	1.4 ± 1.2	NSS		
Ru-103	Distant	<MDC	<MDC	NSS	3,000,000	1
	Boundary	<MDC	1.6 ± 1.4	NSS		
Ru-106	Distant	<MDC	<MDC	NSS	200,000	10
	Boundary	<MDC	9 ± 6	NSS		
Sb-125	Distant	<MDC	<MDC	NSS	900,000	6
	Boundary	<MDC	3 ± 2	NSS		
Sr-90	Distant	<MDC	<MDC	NSS	30,000	0.6
	Boundary	<MDC	0.20 ± 0.18	0.13 ± 0.06		
		Concentration (10^{-18} μCi/mL)				
Am-241	Distant	<MDC	9 ± 6	NSS	200,000	6
	Boundary	<MDC	<MDC	NSS		
Pu-239/240	Distant	<MDC	5 ± 4	1.9 ± 1.3	60,000	6
	Boundary	<MDC	6 ± 4	1.8 ± 1.7		

a. Distant stations are Blackfoot, Craters of the Moon, and Idaho Falls. Boundary stations are Arco, Atomic City, FAA Tower, Howe, Monteview, Mud Lake, and Reno Ranch.

b. Single quarterly composite sample analytical results $\pm 2\sigma$, decay corrected assuming a constant concentration and buildup during the sampling period. See Appendix C.

c. Arithmetic mean with the 95% confidence interval for the mean. See Appendix C.

d. Concentration guides are based on *DOE Order 5480.1A*, Chapter XI guides for release to an uncontrolled area.

e. The minimum detectable concentrations (MDC) are approximate and are calculated for typical values for airflow volume, counting time, radionuclide composition of the sample and time lapsed between collection and analysis. These values may vary slightly for actual samples.

f. Below minimum detectable concentration.

g. Mean is not statistically significant (NSS), or zero is included within the 95% confidence interval for the mean. See Appendix C.

The New Waste Calcining Facility (NWCF) at the Idaho Chemical Processing Plant (ICPP) is operating under a variance from the State of Idaho for opacity requirements. The variance is for visible emissions due to nitrogen dioxide in the plume from the main stack at ICPP. Visual determination of opacity is routinely made twice a week. All of these observations for 1983 were below the 60% opacity specified in the variance.

Water Sampling

The Snake River Plain aquifer which lies beneath the INEL Site serves as the primary source for drinking water and irrigation of crops in the Snake River Basin. Onsite and offsite water samples are collected routinely to monitor for movement of waste substances, both radioactive and nonradioactive, through the aquifer.

Offsite water samples are collected semiannually from drinking water production wells and from the Snake River. Water is also collected at three U. S. Geological Survey (USGS) wells near the southern Site boundary. See Figure 6. All radioactivity detected in offsite water samples is reported and evaluated in this section. Gross alpha, gross beta, and tritium analyses are routinely performed on the water samples. For gross alpha analysis, a portion of the sample is evaporated on a stainless steel planchet and counted with a scintillation counter system. Another portion is evaporated and counted for gross beta activity in a low-background beta counter. Tritium concentrations are determined with a liquid scintillation counter. The minimum detectable concentrations for gross alpha, gross beta, and tritium are 3×10^{-9}, 5×10^{-9}, and 4×10^{-7} μCi/mL, or about 10, 20, and 0.01%, respectively, of DOE concentration guides for an uncontrolled offsite area. These minimum detectable concentrations are also 20, 10, and 2%,

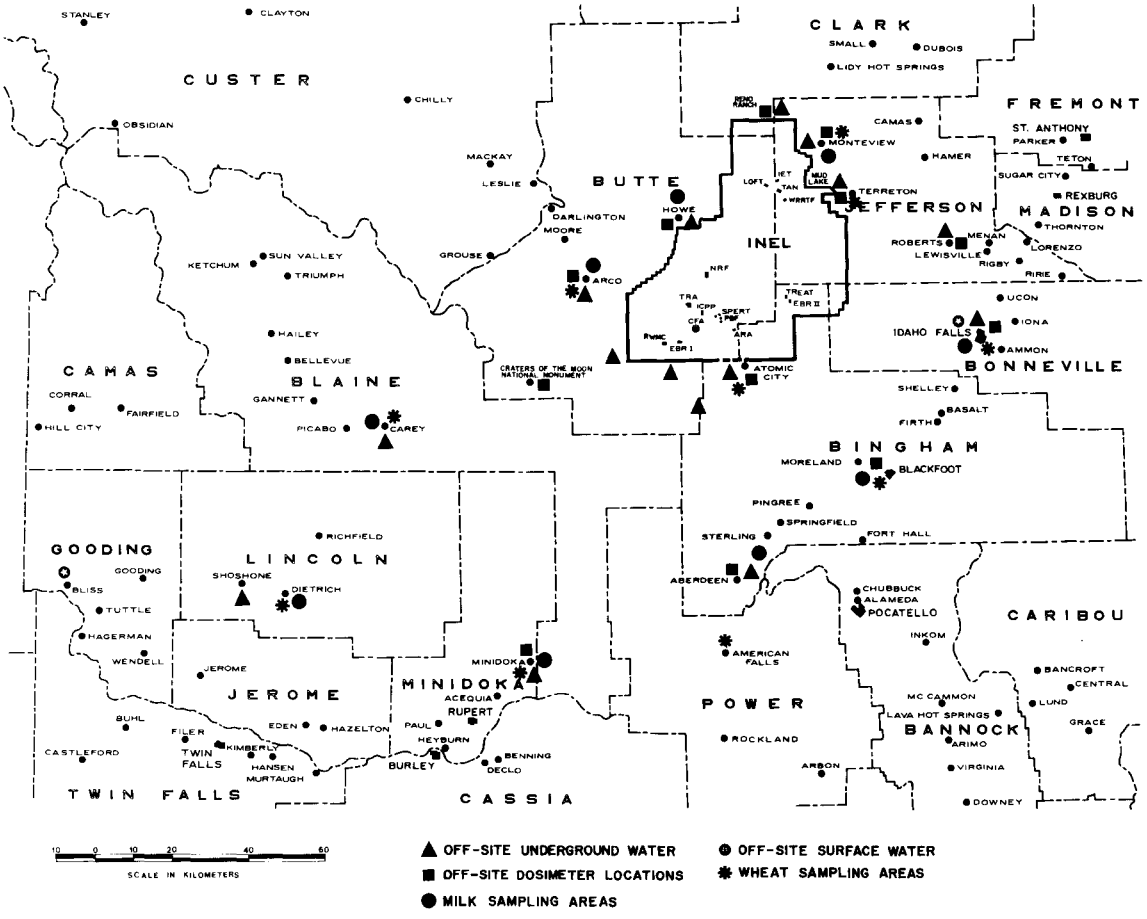

Figure 6. Off-site water, milk, and wheat sampling locations and environmental dosimeter locations.

respectively, of regulations for community drinking water listed by the U. S. Environmental Protection Agency in 1983.

Only two of the 36 offsite water samples, both from Arco, contained gross alpha activity in 1983. Activity in the two samples was near the minimum detectable concentration at $4 \pm 2 \times 10^{-9}$ and $5 \pm 4 \times 10^{-9}$ μCi/mL. Gross beta activity was reported in only three of the 36 water samples, one each from Reno Ranch, USGS Well #11, and the Snake River at Bliss, Idaho. Activities in these three samples were all near the minimum detectable concentration and ranged from $5 \pm 4 \times 10^{-9}$ to $7 \pm 6 \times 10^{-9}$ μCi/mL. Annual averages for gross alpha and gross beta activity at all locations were below the EPA community drinking water standards.

Natural radioactivity is found elsewhere in the Snake River Plain groundwater in areas upgradient, parallel to, and far distant from the INEL Site. Reno Ranch lies upgradient to groundwater flow beneath the INEL;[3] therefore, any radioactivity reported there is not due to Site operations. Arco, USGS Well #11, and Bliss, Idaho are located to the south or west of the Site boundary, but are all well beyond the leading edge of any Site waste plumes.[4] Therefore, it is most likely that the radioactivity reported in those samples is due to either statistical variations (see Appendix C) or to natural radionuclides from soil and rock. One sample from each of the three offsite USGS wells was submitted for gamma spectrometry, but no manmade gamma-emitting radionuclides were detected.

Most of the onsite water sampling is conducted by the U. S. Geological Survey. Tritium was detected in onsite wells near the southern Site boundary during 1983 but was not detected in offsite wells south of the boundary. This indicates that the tritium waste plume has probably reached the Site boundary, but has not traveled very far offsite if at all. Iodine-129 which has a less extensive plume than tritium, is detectable about 6 km (3.7 mi) inside the nearest Site boundary.[5] Strontium-90 analyses were above the minimum detectable concentration only for those samples collected within 3 km (1.9 mi) of the release point at the ICPP disposal well, or approximately 10 km (6.2 mi) inside the nearest Site boundary.[5] The estimated minimum detectable concentrations for Sr-90 and I-129 are about 5×10^{-9} and 1×10^{-10} μCi/mL, respectively, or about 2 and 0.2% of the concentration guides for an uncontrolled area. Cesium and actinides are even less mobile in the aquifer than strontium.

Nonradiological wastes in the aquifer are determined by measuring the specific conductance and the chloride, sodium, nitrate, and total chromium content of the water. (See references 4 and 5.) All of these waste products were at background levels at least 3 km (1.9 mi) inside the nearest Site boundary, indicating that INEL groundwater nonradiological waste plumes had not migrated offsite by the end of 1983.

Foodstuff Sampling

Milk, wheat, and leafy garden lettuce are sampled routinely since they are part of the typical American diet. These three foodstuffs could be pathways to the public for radionuclides from nuclear weapons fallout or from Site operations. Boundary areas are compared to distant areas to assess possible impacts from Site operations. Milk and wheat sampling locations are shown in Figure 6. Lettuce was collected at Arco, Atomic City, Blackfoot, Carey, Howe, Idaho Falls, Mud Lake, and Pocatello, Idaho.

A total of 154 milk samples were collected from dairies around the Site. Samples are collected monthly, except in Idaho Falls where a sample is collected weekly. An exception was Reno Ranch from which only six samples were collected because the family cow was dry part of the year. All milk samples are passed through anion exchange resins which are analyzed for I-131 by gamma spectrometry. Milk from each location is analyzed for Sr-90 and tritium once during each year. In addition, three September samples, one each from Carey, Idaho Falls, and Mud Lake, Idaho are submitted for I-129 analysis.

In 1983 eight milk samples from offsite areas (five distant and three boundary) contained I-131 in concentrations near the minimum detectable concentration. The five concentrations from distant areas ranged from $0.7 \pm 0.6 \times 10^{-9}$ μCi/mL for two samples from Idaho Falls to $1.1 \pm 0.6 \times 10^{-9}$ μCi/mL for the area south of Idaho Falls. The three boundary area samples included one from Howe at $0.7 \pm 0.6 \times 10^{-9}$ μCi/mL, and two from Arco at $0.7 \pm 0.6 \times 10^{-9}$ and $1.3 \pm 0.6 \times 10^{-9}$ μCi/mL. All eight concentrations are well

below health protection guides. If it is assumed the I-131 was actually present in the samples (see Appendix C), consideration of meteorological and Site operations information leads to the conclusion that the detected activity was not due to Site activities.

Three of the six milk samples from distant areas contained Sr-90 at detectable concentrations ranging from $1.7 \pm 1.4 \times 10^{-9}$ to $2.7 \pm 1.6 \times 10^{-9}$ μCi/mL. The distant group mean concentration was 1.7×10^{-9} μCi/mL. This mean concentration is consistent with the trend of Sr-90 levels in Idaho Falls milk samples reported by the Environmental Protection Agency for previous years. Two of the four samples from Site boundary areas also had detectable concentrations of Sr-90 at $1.6 \pm 1.4 \times 10^{-9}$ and $1.9 \pm 1.8 \times 10^{-9}$ μCi/mL. The group mean of 1.4×10^{-9} μCi/mL is not statistically different from the background mean concentration. The origin of the nuclide in the samples from the five areas was probably worldwide fallout. Iodine-129 analyses of the three September samples were not completed at the time of this report, but will be reported next year.

Wheat and leafy garden lettuce sampling results are shown in Table IV. The lettuce was washed with water to remove the obvious dirt, then dried and weighed. Lettuce samples were analyzed for Sr-90

TABLE IV
RADIONUCLIDE CONCENTRATIONS IN WHEAT AND LETTUCE (1983)

			Concentration[a] (10^{-9} μCi/g dry wt)			
			Wheat		Garden Lettuce	
	Group	Sample Location	Sr-90	Cs-137	Sr-90	Cs-137
∿Minimum Detectable Concentration	—	—	4	4	80	200
	Distant	American Falls	10 ± 4	<MDC[b]	NA[c]	NA
		Blackfoot	10 ± 4	<MDC	200 ± 120	<MDC
		Carey	7 ± 4	<MDC	380 ± 240	<MDC
		Dietrich	24 ± 2	<MDC	NA	NA
		Idaho Falls	31 ± 6	<MDC	<MDC	<MDC
		Minidoka	9 ± 4	<MDC	NA	NA
		Pocatello	NA	NA	100 ± 40	<MDC
		Average[d]	15 ± 10	2.2 ± 0.8	NSS[e]	NSS
	Boundary	Arco	5 ± 3	<MDC	210 ± 80	<MDC
		Atomic City	11 ± 4	<MDC	240 ± 60	<MDC
		Howe	NA	NA	120 ± 40	240 ± 190
		Monteview	7 ± 4	<MDC	NA	NA
		Mud Lake	NA	NA	180 ± 60	<MDC
		Terreton	6 ± 4	4 ± 3	NA	NA
		Average[d]	7 ± 4	NSS	190 ± 80	NSS

a. Analytical result $\pm 2\sigma$. See Appendix C.

b. Below minimum detectable concentration.

c. No analysis or no sample collected.

d. Arithmetic mean with the 95% confidence interval for the mean. See Appendix C.

e. Mean is not statistically significant or 0 is included in the 95% confidence interval for the mean. See Appendix C.

and gamma-emitting radionuclides. Comparison of average concentrations of Sr-90 for boundary and distant communities (background) showed no statistical difference between the two groups. Cesium-137 at a level near the minimum detectable concentration was reported only at Howe at $2.4 \pm 1.9 \times 10^{-7}$ μCi/g.

The wheat was weighed prior to analysis but not washed. All wheat samples were analyzed for Sr-90 and Cs-137, and average concentrations were statistically the same for boundary and distant samples for both nuclides. Cesium-137 was reported only at one boundary area at a concentration of $4 \pm 3 \times 10^{-9}$ μCi/g. It is unlikely that these concentrations are due to Site operations. See Appendix C.

Muscle and liver samples were taken from sheep which had grazed onsite in the southern and northeastern grazing areas (two from each area) and from two sheep from the Blackfoot area which had not grazed near the Site in 1983. Cesium-137 was detected in the muscle samples of one sheep which had grazed onsite at a concentration of $9 \pm 6 \times 10^{-9}$ μCi/g and in liver samples of two sheep which had grazed onsite at concentrations of $1.1 \pm 0.6 \times 10^{-8}$ and $3.1 \pm 1.2 \times 10^{-8}$ μCi/g. These concentrations are comparable to those found in both onsite and offsite sheep in previous years.

Since concentrations of Sr-90 and Cs-137 in foodstuff samples from distant stations were statistically the same as those found in samples from boundary stations, it is assumed that the origin of these radionuclides is worldwide fallout.

Environmental Radiation Measurements

Thermoluminescent dosimeters (TLDs) are used to measure ionizing radiation exposures (beta greater than about 200 keV and gamma greater than 10 keV) at six boundary community locations and six distant community locations. Individual station data are presented, trends over time, if any, are analyzed, and the boundary and distant groups are compared. At each location, a dosimeter containing five individual Harshaw TLD-700 chips (3.18 x 3.18 x 0.89 mm) is placed 1 meter above ground level. The dosimeter at each location is changed semiannually. The measured cumulative exposure for the time period from November 1982 to November 1983 is shown in Table V. For purposes of comparison, annual exposures for 1980 through 1982 are also included for each location. Variations of ionizing radiation between locations are evident but have been consistent at each location for the past four years. The TLDs measure ionizing radiation exposures from natural radioactivity in the air and soil, cosmic radiation from outer space, fallout from nuclear weapons tests, radioactivity from fossil fuel burning, and radioactive effluents from Site operations and other industrial processes.

The mean annual TLD exposures for distant and boundary community locations were 115 and 116 mR, respectively, (110 and 111 mrem). The boundary and distant mean exposures are statistically the same, so there were no significant contributions to ionizing radiation doses at boundary locations from INEL operations.

Table VI summarizes the calculated dose rate an individual receives on the Snake River Plain from various background radiation sources. This dose rate varies from year to year depending on the amount of snow cover. For 1983, the background dose rate was about 140 mrem.

Game Species

Hunting and fishing are not allowed on the Site. Game animals do migrate on and off the Site (see Figure 7), representing a potential, but very small, exposure pathway. Game animals which were killed on Site roads were sampled during 1983. Data were obtained as part of DOE research programs rather than as a part of the routine environmental monitoring program. Many of these programs use the expertise of university faculty and graduate students to assist in the radioecology and ecology research at the INEL Site. These research programs, reported in the scientific literature, supplement the scheduled environmental monitoring reported here.

Muscle and liver tissues from one antelope and one deer were submitted for analysis by gamma spectrometry. No manmade gamma-emitting nuclides were detected in tissues of either animal.

Muscle tissues from two sage grouse which were killed on Site roads during 1983 were submitted for gamma spectrometry. Cs-137 was detected in one bird at a concentration of $6 \pm 4 \times 10^{-9}$ μCi/g, and no gamma-emitting nuclides were found in the tissues of the other bird. In a 1977-79 study of radionuclide concentrations in sage grouse muscle, the average concentration for birds collected from control areas was 3×10^{-7} μCi/g.[8]

No fish were taken from the Big Lost River onsite during 1983.

TABLE V
ENVIRONMENTAL RADIATION EXPOSURES (1980-1983)

		Annual Exposures[a] (mR)			
	Location	1980	1981	1982	1983
Distant Group	Aberdeen	121 ± 6	120 ± 4	116 ± 7	112 ± 4
	Blackfoot	118 ± 8	119 ± 4	117 ± 6	119 ± 6
	Craters of the Moon	116 ± 6	123 ± 4	125 ± 4	113 ± 4
	Idaho Falls	102 ± 6	104 ± 4	104 ± 7	111 ± 8
	Minidoka	—[b]	106 ± 4	105 ± 8	104 ± 5
	Roberts	135 ± 9	134 ± 5	130 ± 7	130 ± 5
	Average[c]	118 ± 15	118 ± 12	116 ± 11	115 ± 9
Boundary Group	Arco	120 ± 6	113 ± 4	109 ± 4	110 ± 5
	Atomic City	121 ± 8	128 ± 4	119 ± 4	122 ± 4
	Howe	109 ± 6	113 ± 4	116 ± 3	112 ± 6
	Monteview	112 ± 6	113 ± 4	112 ± 6	114 ± 6
	Mud Lake	126 ± 8	126 ± 6	124 ± 4	122 ± 4
	Reno Ranch	119 ± 7	115 ± 4	110 ± 2	114 ± 5
	Average[c]	118 ± 7	118 ± 7	115 ± 6	116 ± 5

a. Annual exposure $\pm 2\sigma$. See Appendix C.

b. Results were available for only six months of the year at this location.

c. Arithmetic mean with the 95% confidence interval for the mean. See Appendix C.

TABLE VI
BACKGROUND RADIATION DOSE RATE (1983)

Source of Background Dose		Dose Rate (mrem year) Estimated[a]	Measured (TLD)[b]
External	Terrestrial	66	—
	Cosmic (ionizing)	43	—
	Subtotal	109	110
	Cosmic (neutron)	6	—
Internal	K-40 and others	27	—
	Total	142	—

a. Doses are estimated from charts and tables in NCRP Report No. 45.[6] Doses are not strictly additive since some are expressed as exposure in air and others are tissue doses.

b. For conversion from mR in air to mrem in tissue, f factor was 0.96, estimated from Johns and Cunningham.[7]

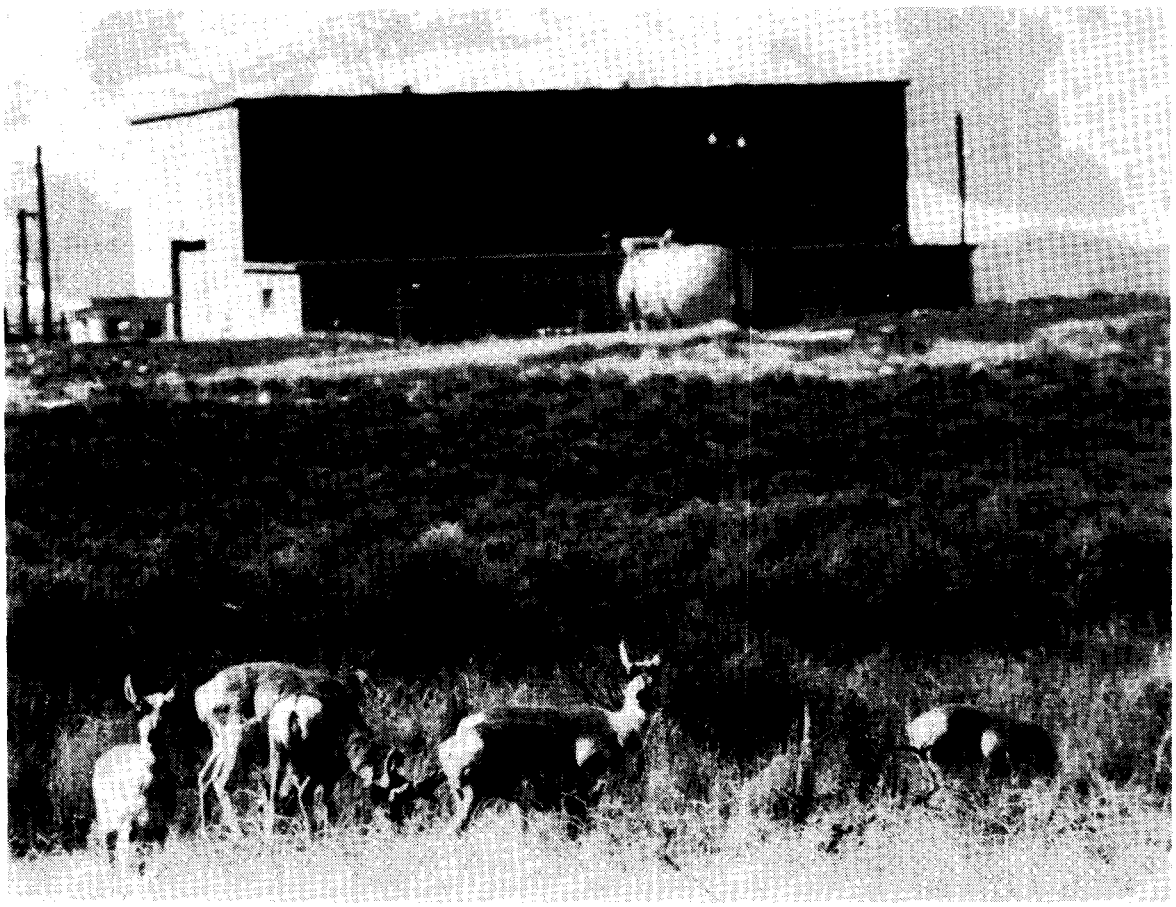

Figure 7. Antelope grazing near the Power Burst Facility on the INEL Site.

RADIOLOGICAL IMPACT OF INEL SITE OPERATIONS

General

The radiological impact of Site operations on the resident public surrounding the Site was too small to be measured by the routine monitoring program. Therefore, the impact was estimated by calculating:

- The maximum "fence post" or 50-year Site boundary dose commitment
- The maximum potential 50-year dose commitment to the nearest individual residing offsite
- The maximum potential 50-year dose commitment to a member of a population group
- The potential 50-year population dose commitment which could have been received by the public within an 80-km (50-mi) radius of the operations center of the Site [Test Reactor Area (TRA) and Idaho Chemical Processing Plant (ICPP)].

For simplicity, the term "50-year dose commitment" calculated for Site releases will be used interchangeably with the term "dose" in the following sections.

The possible exposure pathways by which radioactive materials from Site operations could be transported to offsite environs are shown diagrammatically in Figure 8. Atmospheric transport is the principal potential exposure pathway from the Site. There are no surface streams flowing from onsite to offsite locations. The leading edge of the tritium plume, the most mobile low-level radioactive waste in the aquifer, reached the Site boundary in 1983. Tritium from the Site has not been detected in any offsite wells.

Several indirect exposure pathways are being studied at the Site to determine their effect, if any, on the highest possible dose that could have been received by a member of the public. The principal indirect exposure pathway involves the eating of game species that have spent some time on the Site. Some radioactivity can be present in game species depending upon the length of residence onsite, the time elapsed after migration from the Site and the metabolism of the animal. Conservative estimates of the potential dose to a person consuming meat from different game animals is described in the section "Maximum Individual Dose Commitment."

The monitoring data presented in the previous sections indicated that at offsite sampling locations no particulate radioactivity in the air from Site operations was discernible from the pre-existing levels due to natural and fallout radioactivity. As mentioned in the section on air sampling, noble gases are not sampled by the air monitoring system. Because of these limitations, an estimate of the radiological impact of Site operations on the surrounding region has been made by using the known amounts of various radionuclides released during 1983 from Site facilities and a meteorological model for estimating the concentrations at selected locations in the vicinity. A summary of the radionuclides released to the atmosphere from Site facilities is shown in Table VII. Due to radioactive decay of the short-lived radionuclides, the activity that would reach offsite areas is less than the 12,000 Ci indicated in Table VII. The ICPP and TRA facilities together were the source of about 98% of the total radioactivity released to the atmosphere. Noble gases comprised about 93% of the total radioactive airborne effluent.

The mesoscale meteorological map (Figure 9) shows the calculated 1983 concentrations normalized to a unit release rate for the INEL Site and vicinity. This map has been prepared by the National Oceanic and Atmospheric Administration (NOAA) at the INEL from data gathered continuously at 26 meteorological stations on and around the Site.[9] To facilitate the display, the dispersion coefficient values are given in whole numbers and must be multiplied by 10^{-9} hr^2/m^3. To obtain the average air concentration (Ci/m^3) for a radionuclide released from TRA or ICPP along any dispersion coefficient isopleth in Figure 9, the value of the 1983 average dispersion coefficient (e.g., 30×10^{-9} hr^2/m^3) was multiplied by the number of curies of the radionuclide released during 1983 and was divided by the number of hours in a year squared (7.67×10^7). Logarithmic interpolation between isopleths was used to obtain concentrations at other points.

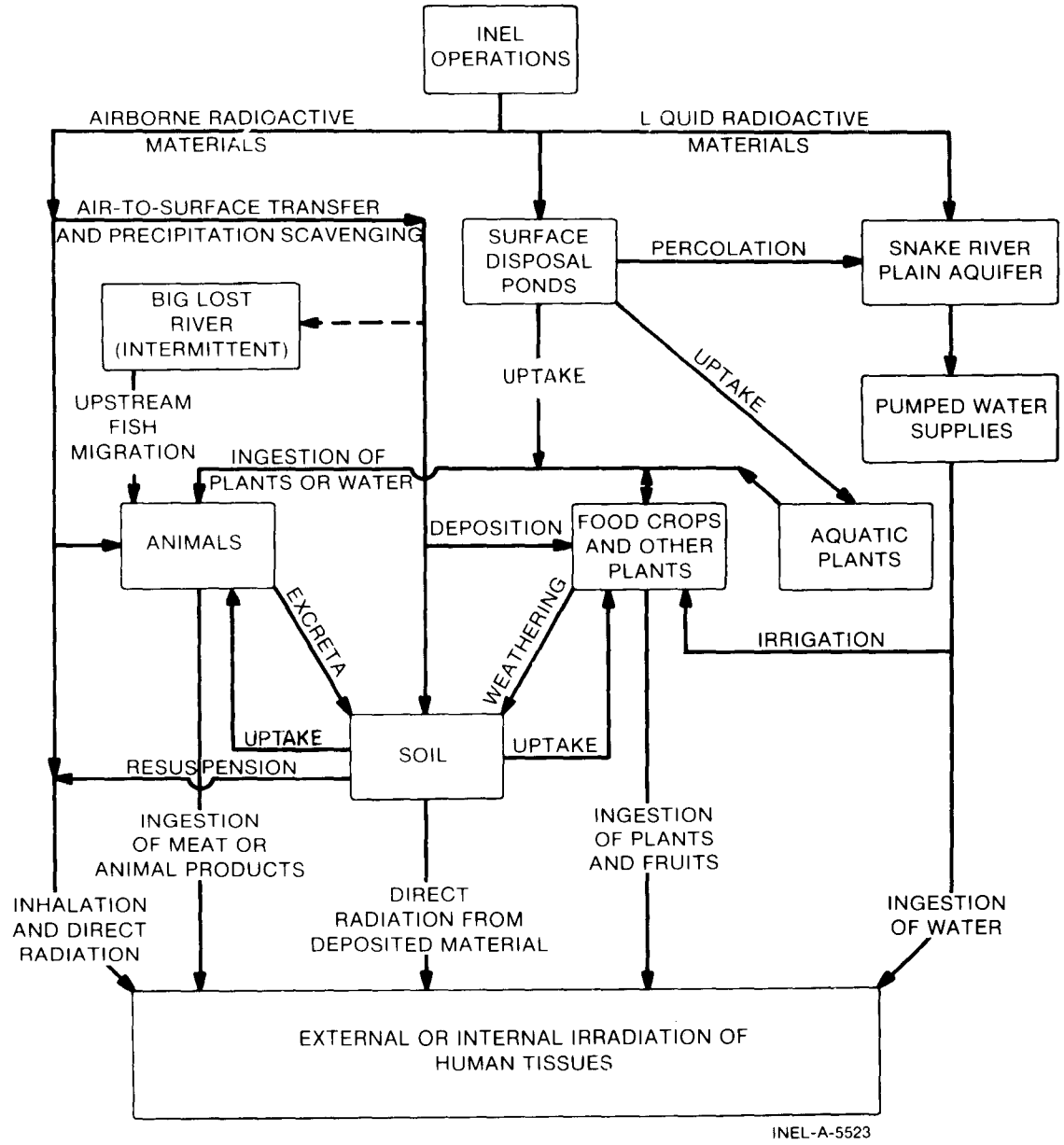

Figure 8. Possible exposure pathways of the INEL Site radioactive materials to humans within 80 km (50 mi) of ICPP and TRA.

TABLE VII
RADIONUCLIDE COMPOSITION OF AIRBORNE EFFLUENTS (1983)

			Airborne Effluent (Ci)[a]			
	Radionuclide	Half-Life	ANL-W	ICPP	TRA	Total[b]
Noble Gases	Kr-85	10.7 yr	16	3,100	—	3,100
	Xe-138	14.2 min	0.6	—	2,300	2,300
	Ar-41	1.83 hr	130	—	2,100	2,300
	Kr-88	2.84 hr	4.5	—	880	880
	Kr-87	1.27 hr	1.7	—	860	860
	Xe-135	9.09 hr	35	—	820	860
	Xe-135m	15.3 min	0.5	—	420	420
	Xe-133	5.25 da	70	—	260	330
	Kr-85m	4.48 hr	3.2	—	250	250
Particulates	Ba-139	1.39 hr	—	—	150	150
	Cs-138	32.2 min	—	—	16	16
	Rb-88	17.7 min	—	—	11	11
	Br-82	1.47 da	0.9	—	—	0.9
	Cs-137	30.2 yr	—	4.3×10^{-3}	2.2×10^{-5}	4.3×10^{-3}
	Sr-90/Y-90[c]	28.6 yr	—	1.2×10^{-3}	4.6×10^{-7}	1.3×10^{-3}
	Pu-238	87.7 yr	—	1.5×10^{-3}	—	1.5×10^{-3}
	Ru-106	369 da	—	8.2×10^{-4}	—	8.2×10^{-4}
	Pu-239/240	2.4×10^{4} yr	—	2.0×10^{-4}	—	2.0×10^{-4}
H-3, C-14, and Iodine Isotopes	H-3	12.3 yr	2.	660	—	660
	I-129	1.6×10^{7} yr	—	0.3	—	0.3
	C-14	5.7×10^{3}	—	4.1×10^{-2}	—	0.2
All Others Total			2.2×10^{-6}	3.1×10^{-3}	4.8×10^{-6}	3.1×10^{-3}
Total			260	3,700	8,100	12,000

a. Radioactivity listed in 1983 Waste Management Information System Report.[1] Values are not corrected for decay after release. Data are preliminary. Revised data were obtained for the ICPP (personnel communication).

b. Totals include small amounts from facilities not listed.

c. Parent-daughter equilibrium assumed.

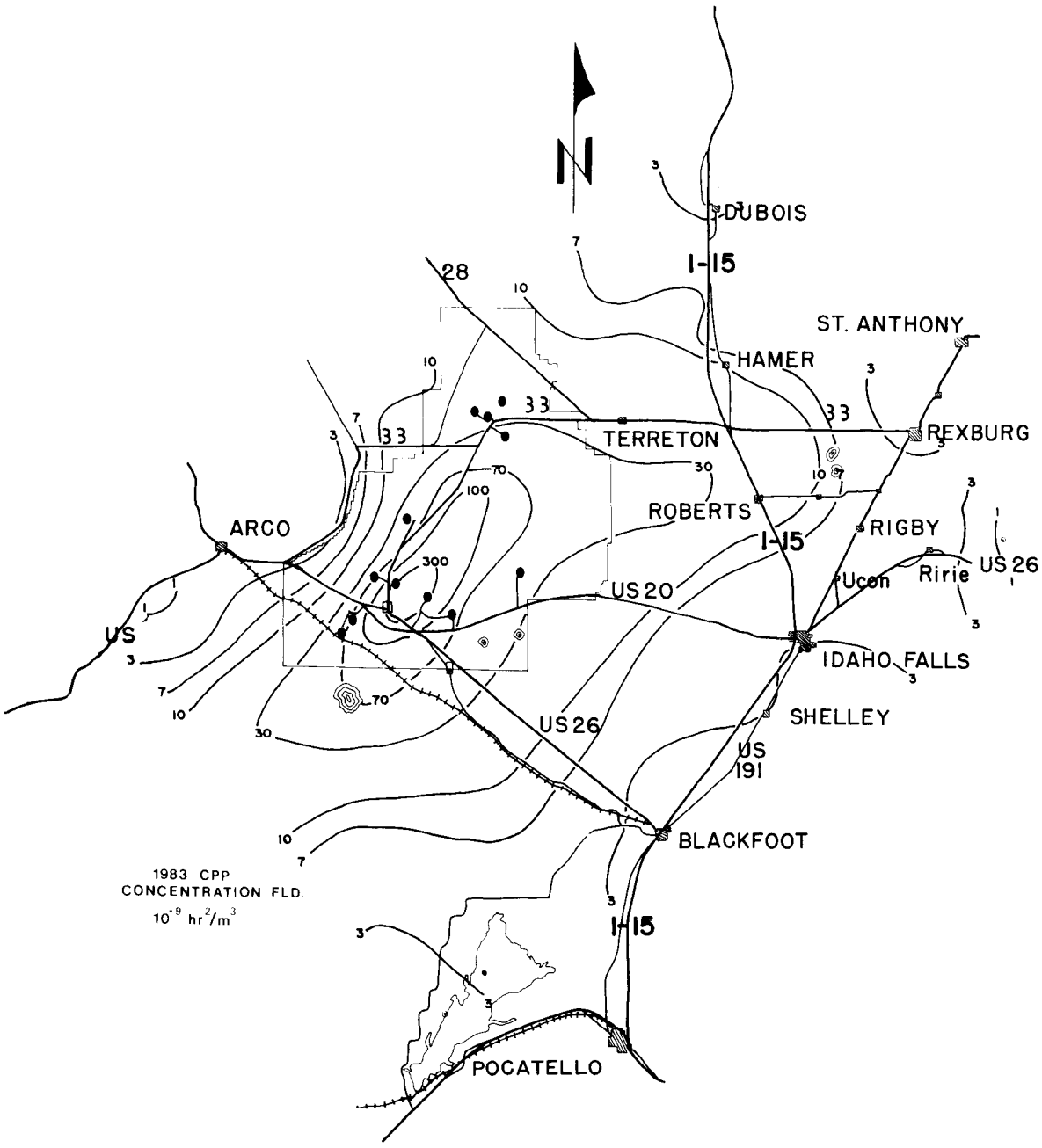

Figure 9. 1983 average mesoscale dispersion isopleths of air concentrations at ground level, normalized to unit release rate.

Maximum Site Boundary Whole-Body Dose Commitment

The maximum hypothetical 50-year whole-body dose commitment to an adult from inhalation of air, submersion in air, and radiation due to deposition of particulates on soil was calculated assuming that an individual resided continuously for a year at the point of maximum radionuclide concentration just outside the Site boundary (fence post dose). The calculation was based on data presented in Table VII and Figure 9. The maximum offsite concentration occurred along the southern Site boundary between the isopleths labeled "70" and "100" in Figure 9. The dispersion coefficient used for this point is 85 x 10^{-9} hr^2/m^3. The whole-body dose from each radionuclide in Table VIII was computed using the dose conversion factors given in References 10 and 11. The maximum hypothetical whole-body dose estimated for an adult from Site airborne effluent is 0.02 mrem for 1983. About

TABLE VIII
MAXIMUM SITE BOUNDARY WHOLE-BODY DOSE COMMITMENT (1983)

Radionuclide[a]	Maximum Offsite Concentration[b] (μCi/mL)	Maximum Whole-Body Dose[c] (mrem)
Kr-88	6.4×10^{-13}	0.0091
Ar-41	1.2×10^{-12}	0.0076
Cs-137	4.7×10^{-18}	0.0024
Kr-87	3.8×10^{-13}	0.0016
Xe-135	8.0×10^{-13}	0.00090
H-3	7.3×10^{-13}	0.00037
Xe-138	1.7×10^{-14}	0.00030
Sb-125	3.2×10^{-18}	0.00017
Kr-85m	2.1×10^{-13}	0.00016
Total	—	0.023

a. Table includes only radionuclides which contribute a dose of 0.0001 mrem or more.

b. Estimate of radioactive decay obtained by using the 1983 average windspeed from 355-005° of 8300 m/hr and a distance of 14,200 m from TRA-ICCP to point of maximum offsite concentration.

c. Whole-body dose estimated using parameters given in Kocher[10] and in ICRP-30.[11] Doses are 50-year dose commitments.

86% of that computed dose was due to noble gases and particulates with half-lives of less than 10 hours. This dose is 0.004% of the radiation protection standard for exposure to an individual in an uncontrolled area (*DOE Order 5480.1A*, Chapter XI). Calculations were also made of doses to several body organs and tissues (lung, liver, kidney, red bone marrow, bone surfaces, thyroid and skin). The largest dose to an organ or tissue was 0.08 mrem to the skin which is 0.005% of the radiation protection standard for the skin.

The whole-body dose (0.02 mrem) resulting from Site operations is very small compared to the estimated 140 mrem received from cosmic and terrestrial radiation during 1983. For interest, it may also be compared to the approximately 36 mrem from medical and radiological diagnostic procedures, to the estimated 25 mrem received each year from natural radionuclides in the body, to about 3.5 mrem received during a 5-hour transcontinental jet flight, or to the 0.05 to 0.01 mrem received annually by the average television viewer.[12]

Maximum Individual Dose Commitment

As indicated in Figure 9, Atomic City was the location nearest to the Site boundary where people actually reside and thus represents the point of the greatest probable 50-year dose commitment from Site operations. Using 50×10^{-9} hr^2/m^3 as the dispersion coefficient for Atomic City and allowing for radioactive decay during the transit of the radionuclides to Atomic City, the potential individual dose from inhalation, submersion, and deposition was calculated to be 0.014 mrem. This dose is about 0.003% of the radiation protection standard for exposure to an individual at points of maximum probable exposure (DOE Order 5480.1A, Chapter XI). No allowance was made for shielding by housing or residence time in the community.

Potential dose to an individual from ingestion of meat from game animals continues to be investigated. One group of studies involves the calculation of potential doses to individuals who might eat ducks which reside briefly upon a liquid waste pond

used for the disposal of low-level reactor effluents (Figure 10). The average whole body dose from consumption of a contaminated duck is 10 mrem.[13] This value is based on the assumption that the duck would be killed and eaten immediately after leaving the pond. Normally, immediate killing of the duck would not occur, so a lower dose would be more realistic due to biological elimination of the radioactivity. Because only about one duck in 4000 passing through this area has a chance of becoming contaminated, the probability of receiving this dose is further reduced.[14]

The highest estimated potential dose to a person eating the entire muscle mass of a sage grouse which summered near the TRA-ICPP area was 2 mrem.[8] Sage grouse which summered in other Site areas or offsite were calculated to provide estimated doses of 0.01 to 0.04 mrem.

The calculated maximum radiation dose to a person eating the muscle tissue of one mourning dove was 0.3 mrem. The average dose to people consuming doves migrating from the Site areas was 0.01 mrem which is the same as for control birds collected far from the Site.[15]

A conservative estimate of the maximum dose which could have been received by a single individual eating the entire muscle and liver mass of an antelope (collected on the INEL after August 1975) with the highest levels of radionuclides was less than 0.2 mrem.[16]

Maximum Dose Commitment to a Member of a Population Group

Atomic City is the population group nearest to the point of maximum exposure on the Site boundary. Therefore, each resident of this community would have the same potential 50-year dose commitment as calculated in the section above. However, this would be compared to the standard for a suitable sample of the exposed population. The 0.014 mrem dose is about 0.008% of that standard.

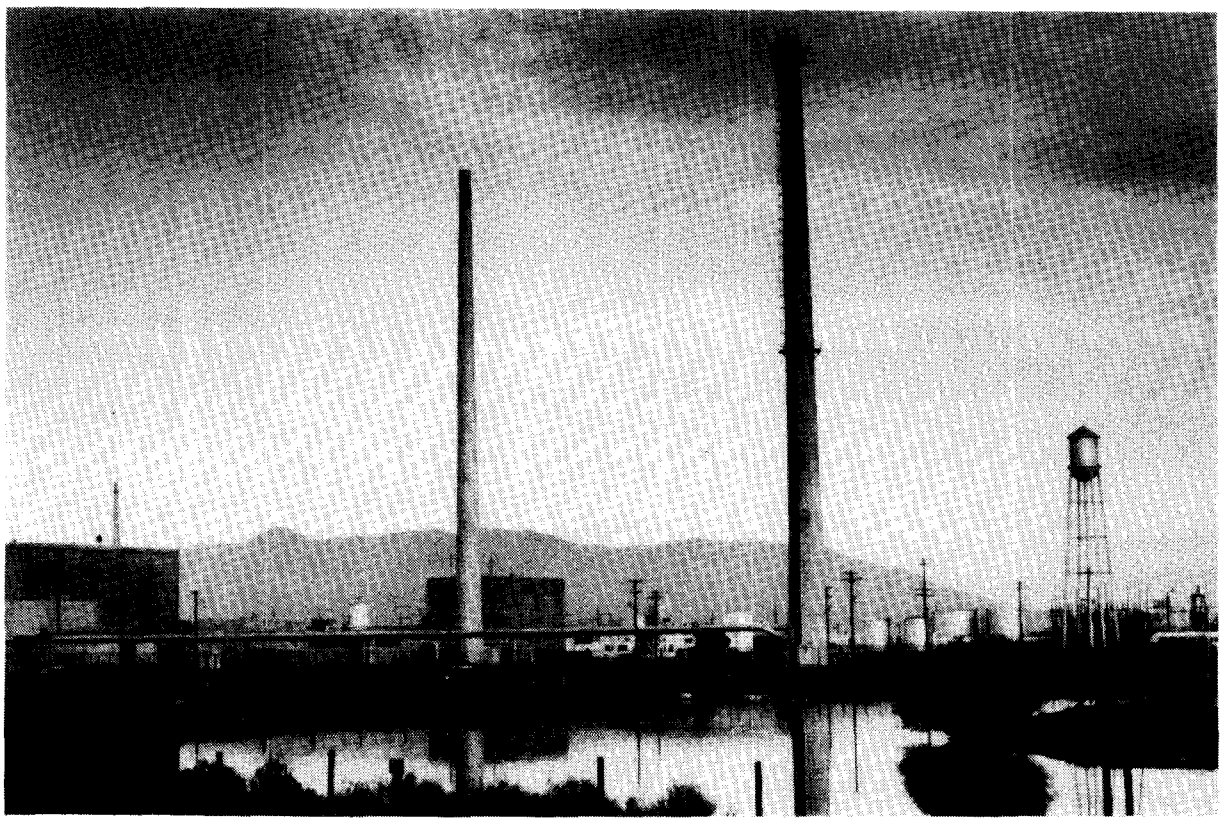

Figure 10. TRA low-level waste disposal pond on the INEL Site.

TABLE IX
80-KILOMETER POPULATION DOSE COMMITMENT (1983)

Census Division	Dispersion Coefficient[a]	Population[b] 1983	Population Dose[c] (man-rem)
Aberdeen	3×10^{-9}	2,850	0.0020
Arco	2×10^{-9}	2,900	0.0014
Atomic City (city)	50×10^{-9}	34	0.0004
Atomic City (division)	5×10^{-9}	2,230	0.0026
Blackfoot	2×10^{-9}	12,200	0.0057
Carey (part)	1×10^{-9}	100	0.00002
Clark, West (part)	7×10^{-9}	130	0.0002
Firth	2×10^{-9}	3,020	0.0014
Fort Hall (part)	2×10^{-9}	4,150	0.0020
Hamer	25×10^{-9}	2,340	0.014
Howe	9×10^{-9}	450	0.0010
Idaho Falls	3×10^{-9}	60,800	0.043
Idaho Falls, West	5×10^{-9}	1,660	0.0020
Lewisville-Menan	10×10^{-9}	3,180	0.0075
Mackay	1×10^{-9}	650	0.0002
Moreland	3×10^{-9}	7,770	0.0055
Roberts	15×10^{-9}	1,330	0.0063
Shelley	3×10^{-9}	5,800	0.0041
Ucon (part)	6×10^{-9}	3,000	0.0042
Totals	—	114,594	0.10

a. Coefficient, obtained from Figure 9, is the 1983 average concentration normalized to unit release rate (hr^2/m^3). The value selected represents an estimated average based on the location of population centers in the census division.

b. Population for each division is based upon the 1980 Advance Census Reports for Idaho. The population of the City of Idaho Falls with the Idaho Falls census division has been increased by 2.7% per year based upon projections by the Chamber of Commerce. Estimates are also made when only part of a division is located within the 80-km radius.

c. This population dose does not include radioactive decay beyond 14.2 km.

80-Kilometer Population Dose Commitment

An estimate of the maximum 50-year whole-body dose commitment from inhalation, submersion, and deposition which could have been received by all members of the public within an 80-km (50-mi) radius of the TRA-ICPP complex was made by summing the potential individual doses to the people of each census division within the 80-km (50-mi) radius. The dose to an individual of a particular division is a fraction of the maximum individual dose (fence post dose) calculated in a previous section. The fraction is obtained by taking the ratio of the estimated dispersion coefficient for each census division from Figure 9 to the dispersion coefficient of 85×10^{-9} hr^2/m^3 which was used to calculate the maximum individual dose. The potential dose to the population of the division is the product of the potential dose to each resident times the division population. The calculation is conservative since radioactive decay of the isotopes was not calculated during transport over distances greater than the 14 km (9 mi) from the TRA-ICPP midpoint to the southern Site boundary. Idaho Falls, for example, is about 66 km (41 mi) from TRA-ICPP.

The 80-km (50-mi) population dose was the sum of population doses for the various census divisions. The results are summarized in Table IX. The estimated potential population dose was 0.10 man-rem to a population of 114,600. When compared with an approximate population dose of 16,000 man-rem from natural background, this represents an increase of only about 0.0006%. The dose of 0.10 man-rem can also be compared to the following estimated whole-body population doses for the same size population: 4,100 man-rem for medical and radiological diagnostic procedures and 120 man-rem for a group of three common sources of miscellaneous radiation—air transport, self-luminescent consumer products, and television viewing.[12]

The contribution of indirect exposure pathways to the population dose has not been considered because of uncertainties regarding the number of people exposed, the small probability of obtaining game animals migrating from the Site during hunting season, and the levels of different radionuclides in various animals. The dose contribution from these indirect exposure pathways would realistically be less than the dose from inhalation of air, submersion in air, and deposition on soil.

ENVIRONMENTAL STANDARDS AND REGULATIONS

The following environmental standards and regulations are applicable at the INEL Site boundary.

"Requirements for Radiation Protection," Chapter XI, *DOE Order 5480.1A*, August 1981.

U.S. Federal Radiation Council, *Background Material for the Development of Radiation Protection Standard*, Report No. 1, (1960) and Report No. 2 (1961), Superintendent of Documents, U.S. Government Printing Office, Washington, DC

U.S. Environmental Protection Agency, *National Primary and Secondary Ambient Air Quality Standards*, 40 CFR 50, 1983.

U.S. Environmental Protection Agency, *Drinking Water Regulations*, 40 CFR 141, 1983.

Department of Health and Welfare, State of Idaho, *Rules and Regulations for the Control of Air Pollution in Idaho*, 1972 as amended.

Idaho Department of Health and Welfare, State of Idaho, *Idaho Regulations for Public Drinking Water Systems*, 1977.

The principal standards and guides for releases of radionuclides at the INEL are those of *DOE Order 5480.1A*, Chapter XI, dated August 13, 1981 entitled "Requirements for Radiation Protection." Radiation protection standards and selected radioactivity concentration guides from Chapter XI are listed in Tables X and XI. The most restrictive guide is listed when there is a difference between soluble and insoluble chemical forms. These listed guides are identical to those in the *Idaho Radiation Control Regulations*, Radiation Control Section, State of Idaho, 1982.

Ambient air quality standards are shown in Table XII. Water Quality standards are dependent on the type of drinking water system sampled. For public community drinking water systems, Table XIII is a partial list of maximum contaminant levels set by the U.S. Environmental Protection Agency. State of Idaho regulations are the same for those contaminants listed here.

TABLE X
RADIATION PROTECTION STANDARDS *DOE Order 5480.1A*, Chapter XI

Standards	Annual Dose Commitment (mrem/yr)	
	Whole Body, Gonads, or Bone Marrow	Other Organs
Individuals at points of maximum probable exposure	500	1500
Suitable sample of the exposed population	170	500

TABLE XI
RADIOACTIVITY CONCENTRATION GUIDES FOR EFFLUENT RELEASES TO UNCONTROLLED AREAS *DOE ORDER 5480.1A*, CHAPTER XI

Radionuclide	Concentration Guide (μCi/mL) In Air	In Water
Gross Alpha	2×10^{-14}	3×10^{-8}
Gross Beta[a]	1×10^{-12}	3×10^{-8}
Am-241	2×10^{-13}	4×10^{-6}
Sb-125	9×10^{-10}	1×10^{-4}
Ar-41	4×10^{-8}	—
Ba-140	1×10^{-9}	2×10^{-5}
Cs-134	4×10^{-10}	9×10^{-6}
Cs-137	5×10^{-10}	2×10^{-5}
H-3	2×10^{-7}	3×10^{-3}
I-129	2×10^{-11}	6×10^{-8}
I-131	1×10^{-10}	3×10^{-7}
Kr-85	3×10^{-7}	—
Kr-85m	1×10^{-7}	—
Kr-87	2×10^{-8}	—
Kr-88	2×10^{-8}	—
Pu-238	7×10^{-14}	5×10^{-6}
Pu-239	6×10^{-14}	5×10^{-6}
Pu-240	6×10^{-14}	5×10^{-6}
Ru-106	2×10^{-10}	1×10^{-5}
Sr-90	3×10^{-11}	3×10^{-7}
Xe-133	3×10^{-7}	—
Xe-135	1×10^{-7}	—
Xe-138	3×10^{-8}	—

a. Based on the most restrictive beta emitter (Ra-228).

TABLE XII
AMBIENT AIR QUALITY STANDARDS[a]

Pollutant	Type of Standard[a]	Sampling Period	U.S. EPA (μg/m^3)	State of Idaho (μg/m^3)
SO_2	S	3-hour Average	1300	1300
	P	24-hour Average	365	365
	P	Annual Average	80	80
NO_2	S&P	Annual Average	100	100
Total Particulates	S	24-hour Average	150	150
	S	Annual Average	60	60

a. National primary (P) ambient air quality standards define levels of air quality to protect the public health. Secondary (S) ambient air quality standards define levels of air quality to protect the public welfare from any known or anticipated adverse effects of a pollutant.

TABLE XIII
MAXIMUM CONTAMINANT LEVELS FOR PUBLIC COMMUNITY DRINKING WATER SYSTEMS

Gross Alpha	15 pCi/L
Gross Beta	50 pCi/L
Man-made Radionuclides	Concentrations resulting in 4 mrem total body or organ dose equivalent
Tritium[a]	20,000 pCi/L
Strontium-90[a]	8 pCi/L
Nitrate (as N)[b]	10 mg/L
Chromium	0.05 mg/L

a. Based on a 2-L/day drinking water intake.

b. Applies to non-community water systems also.

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APPENDIX A
MAJOR PROGRAMS, LOCATION, GEOLOGY, AND CLIMATOLOGY

The Idaho National Engineering Laboratory Site (INEL) was established in 1949 as the National Reactor Testing Station to provide an isolated station where various kinds of nuclear reactors and support facilities could be built and tested, primarily to demonstrate that nuclear energy could be safely harnessed for generating electricity and other peaceful uses. More nuclear reactors have been built at the INEL Site than at any other location in the world. A total of 52 reactors have been built, of which 15 are operating or operable. The INEL's broad mission is to develop economic energy sources by applying its engineering and scientific expertise to the Department of Energy's (DOE) research and development programs. Major DOE programs currently under way at the INEL Site fall into six categories:

1. Providing test irradiation services from the high-flux test reactor—the Advanced Test Reactor (ATR).

2. Recovering uranium from highly enriched spent fuels and calcining liquid radioactive waste solutions into a solid form for storage at the Idaho Chemical Processing Plant (ICPP).

3. Conducting light-water-cooled reactor safety testing and research at the Loss-of-Fluid Test (LOFT) and the Power Burst Facility (PBF).

4. Operating the Experimental Breeder Reactor No. 2 (EBR-II).

5. Operating the Naval Reactors Facility (NRF).

6. Storing and monitoring solid transuranic wastes.

See Figure A-1 and Table A-1 for the location of INEL Site facilities and an explanation of their acronyms.

The Site is situated on the Upper Snake River Plain in southeastern Idaho at an average elevation of 1500 m (4900 ft). The Site encompasses 2300 km^2 (890 mi^2); it extends 63 airline km (39 mi) from north to south and is about 58 km (36 mi) wide at its broader southern part. The nearest 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, Idaho (see Figure 1 in the main text). With a population of about 1300, Arco is the largest boundary community in the area surrounding the Site. Land immediately beyond the boundaries of the Site is either desert or agricultural land. Most of this nearby farming is concentrated northeast of the Site. Large areas of agricultural land are farmed in the Snake River Valley regions which are more distant from the Site.

The desert plain on which the INEL Site is located, is part of a cool desert shrub biome. Average annual temperature at the Site is 5.6°C (42°F) with extremes of 39°C (103°F) and -44°C (-47°F). Vegetation is typical of a cool desert with sagebrush conspicuous over 80% of the Site. Frequenting the Site are the pronghorn antelope and a few deer, but various kinds of birds, reptiles, and large populations of small mammals are also present. The INEL has been made a National Environmental Research Park (NERP), where scientists from DOE, other federal and state agencies, universities, and private research foundations can study changes caused by man's activities and obtain data for use in making decisions on land use. At present, about 25 different environmental studies are being conducted.

The surface of the plain is a combination of basaltic lava outcroppings and alluvial sedimentary deposits. The sediments range from gravels and sands deposited by streams (as alluvial fans, channel fillings, and deltas), to silts and clays deposited in playas. The subsurface of the plain is principally composed of interbedded basalt flows, lucustrine, and alluvial sedimentary deposits to a depth of about 760 m (2500 ft). The most recent volcanism, 1600 years ago, is evident in the scenic basalt flows at Craters of the Moon National Monument, about 30 km (19 mi) to the southwest of the Site.

Annual precipitation in the Site area has averaged 22 cm (8.5 in.) over the past 15 years. Underlying the desert plain is a natural aquifer in the basaltic

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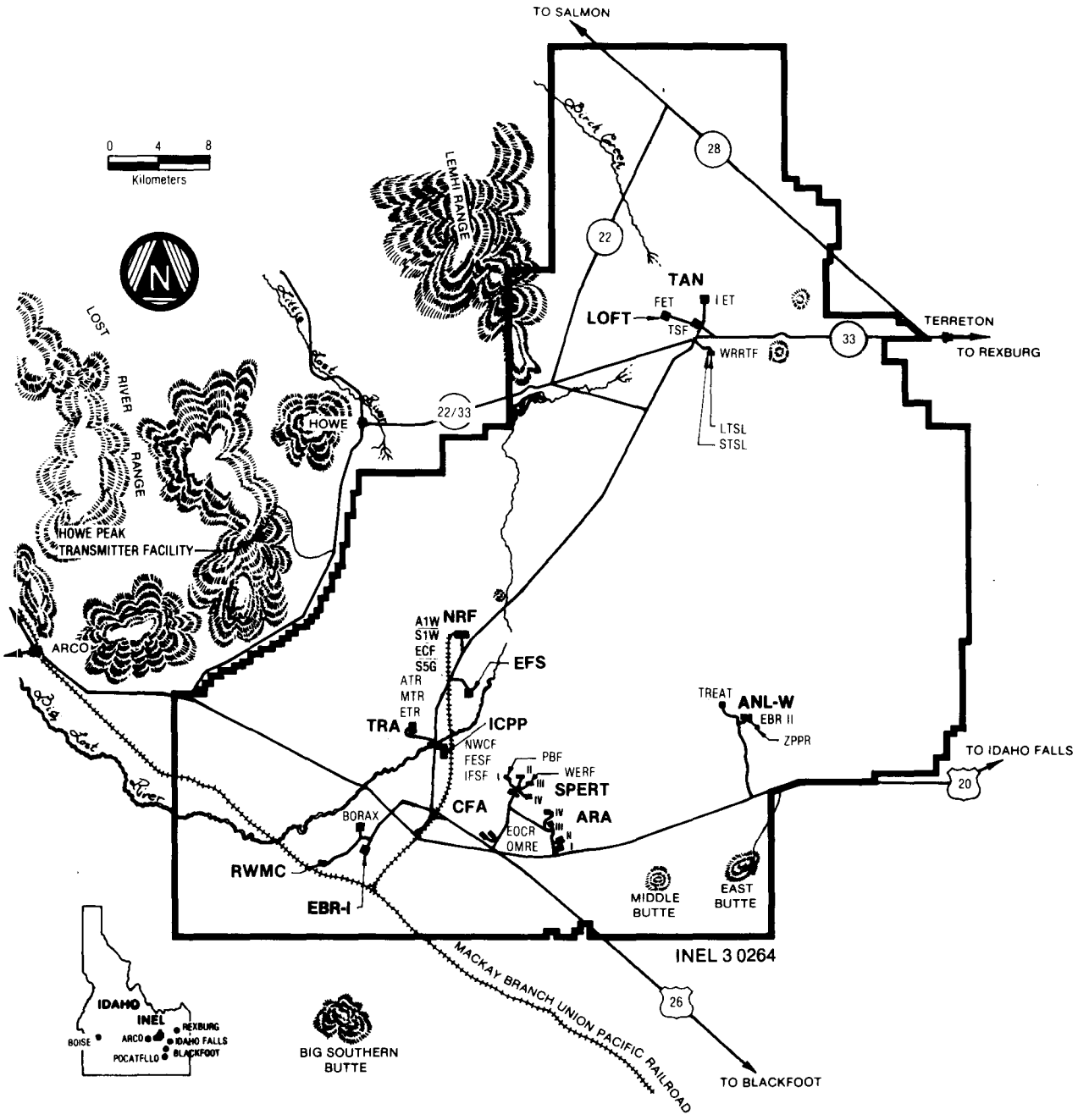

Figure A-1. INEL Site facility locations.

TABLE A-1

TABULATION OF FACILITIES AT THE IDAHO NATIONAL ENGINEERING LABORATORY

Name	Abbreviation	Operating Contractor[a]
Reactors Operating or Operable as of December 1982		
Advanced Reactivity Measurement Facility No. 1[b]	ARMF-I	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
Loss-of-Fluid Test Facility	LOFT	EG&G
Natural Circulation Reactor	S5G	WEC
Power Burst Facility	PBF	EG&G
Submarine Thermal Reactor	S1W (STR)	WEC
Transient Reactor Test Facility	TREAT	ANL
Neutron Radiography Facility[b]	NRAD	ANL
Zero Power Plutonium 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
Materials Test Reactor	MTR	PPCo & INC
Organic Moderated Reactor Experiment	OMRE	AI
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
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
Other Facilities in Use (Continued)		
Computer Science Center (Idaho Falls)	CSC	EG&G
Expended Core Facility	ECF	WEC
Experimental Field Station	EFS	DOE-ID
Field Engineering Test Facility	FET	EG&G
Fuel Element Storage Facility	FESF	ENICO
Hot Fuel Examination Facilities	HFEF	ANL
Hot Pilot Plant	HPP	ENICO
Idaho Chemical Processing Plant	ICPP	ENICO
Idaho Laboratory Facility (Idaho Falls)	ILF	EG&G
Irradiated Fuel Storage Facility	IFSF	ENICO
LOFT Test Support Laboratory	LTSL	EG&G
Naval Reactors Facility	NRF	WEC
New Waste Calcining Facility	NWCF	ENICO
Radioactive Waste Management Complex	RWMC	EG&G
Radiological and Environmental Sciences Laboratory	RESL/ID	DOE-ID
Reactor Training Facility	RTF	EG&G
Semiscale Test Support Laboratory	STSL	EG&G
Standards Calibration Laboratory (CF-698)	--	EG&G
Technical Services Center (CF-688, 689)	TSC	EG&G
Technical Service Facility	TSF	EG&G
Test Area North	TAN	EG&G
Test Reactor Area	TRA	EG&G
Waste Experimental Reduction Facility	WERF	EG&G
Willow Creek Building (Idaho Falls)	WCB	EG&G
Facilities Not Presently in Use		
Coal-Fired Steam Generating Facility	CFSGF	ENICO
Initial Engineering Test Facility	IET	EG&G
Fluorinel and Fuel Storage Facility	FAST	ENICO
Waste Calcining Facility	WCF	ENICO
Major Programs at INEL		
Chemical Processing Program		ENICO
Liquid Metal Fast Breeder Reactor Program		ANL
Naval Propulsion Reactors Program		WEC
Reactor Materials Testing Program		EG&G
Transuranic Waste Management Program		EG&G
Water Reactor Safety Program		EG&G

[a] Operating contractor acronyms: Atomics 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), Westinghouse Electric Corporation (WEC).

[b] Zero or low power reactor.

lava rock. The lateral flow of this water is one billion gallons per day. Aquifer water is believed to be supplied by Henry's Fork of the Snake River. Additional water comes from the Big and Little Lost Rivers and Birch Creek, which start in the mountains to the north and west and sink into the porous soils of the Site area. The underground water moves laterally at an average rate between 1.5 to 6 m per day (5 to 20 ft per day) to the south and west, emerging in springs along the Snake River between Milner and Bliss, Idaho. Both aquifer and surface waters of the Snake River Plain are used for irrigation of crops.

Winds are predominantly along the SW-NE axis of the plain with the most frequent and strongest winds from the SW. The NE winds are mostly nocturnal. Spring is the windiest time of the year, and winter has more calm periods and more night-time temperature inversions.

APPENDIX B
QUALITY ASSURANCE

A quality control and assurance program is maintained by the Radiological and Environmental Sciences Laboratory (RESL/ID) to assure consistent and reliable monitoring results. An internal quality control program is maintained by:

- Adherence to written procedures for sample collection and analytical methods[B-1]
- Documentation of program changes
- Routine calibration of instrumentation
- Frequent equipment performance checks for background and counting rates for standards
- Routine yield determinations of radiochemical procedures
- Duplicate samples to determine precision
- Analysis of quality control standards in an appropriate matrix
- Analysis of reagent blanks to verify chemical purity
- Propagation of all random and systematic uncertainties.

The calibration of counting instruments is carefully performed and is traceable to the National Bureau of Standards (NBS). Six times per year tracer solutions are submitted to the RESL/ID for analysis by gamma spectrometry. Comparisons are also made for beta emitters, including Sr-90 and tritium, and for alpha emitters such as Pu-238, Pu-239, and Am-241. The results are reported directly to the NBS. Results have repeatedly demonstrated traceability to the NBS.

In past years RESL/ID has also participated in the quality assurance program administered by the Environmental Measurements Laboratory of the Department of Energy, the American Society for Testing Materials' round robin testing of standard methods, and in intercomparison with the U. S. Environmental Protection Agency in Las Vegas, Nevada.

The ambient nitrogen oxides and sulfur dioxide analyzers undergo a multipoint calibration every six months, or whenever performance checks indicate poor instrument response. Performance checks are made at least every two weeks and include testing the response of the analyzer to purified air and to air with a known concentration between 150 and 260 $\mu g/m^3$ of nitric oxide or sulfur dioxide. Gas standards used for multipoint calibrations and performance checks are designated protocol gases by the U.S. Environmental Protection Agency and are traceable to the National Bureau of Standards.

To verify the quality of the environmental dosimetry program, in addition to the internal quality control program, RESL/ID has participated in five International Environmental Intercomparison Studies, originally organized by the Environmental Measurements Laboratory and the University of Texas School of Public Health. During 1981 the RESL/ID became an organizer replacing the University of Texas. The RESL applied dosimetry section still participates in the program independently from the RESL staff who conduct the intercomparison. The RESL/ID results have been within 10% of the test exposure values.

The calibration source for the environmental dosimetry program was included in the DOE Intercomparison of Radiological Standards in December of 1983 and found to agree with the reference instrument to within 0.2%.

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APPENDIX C
STATISTICAL METHODS

Individual analytical results are given in the report with plus or minus (±) two analytical standard deviations (2σ) where all analytical uncertainties have been properly propagated. Many of the results were less than or equal to 2σ (and, in fact, some were negative) which is considered as meaning that they were below the minimum detectable concentration.

If the result lies in the range of two to three times its estimated analytical uncertainty (2σ to 3σ), detection of the material by the analysis may be questionable due to statistical variations within the group of samples. Analyses with results in this questionable range are published in this report with the understanding that there may be some doubt as to whether the material was actually present. There are many factors which can influence the result to some degree, and these factors are considered and included in the methods used to determine the uncertainty of the measurement. Small factors are not particularly important when the size of the measurement is many times larger than the uncertainty (e.g. 40 ± 2). However, they may become quite important when working near the minimum detectable concentration where the uncertainty in the measurement is nearly equal to the measurement itself, and the lower limit of the range of the measurement approaches zero. For example, 0.8 ± 0.7 means that one could predict with reasonable confidence that the actual value lies between 0.1 and 1.5. However, such a result may not be very reliable if there is a small factor which has not been included in the uncertainty. The true value of the measurement, in that case, may be zero; or the material being measured is not present. Therefore, when analytical results show a measurement very near the minimum detectable concentration, statistical tools, meteorological data, and Site release information are all considered when interpreting and evaluating the results.

If the result exceeds 3σ, there is confidence that the material was detected by the analysis.

Arithmetic means were calculated using actual assay results, regardless of their being above or below the minimum detectable concentration. The uncertainty of the mean, or the 95% confidence interval was determined by multiplying the standard deviation of the mean (also called the standard error of the mean) by the t statistic. Means for which the 95% confidence interval does not include zero were assumed to indicate detectable amounts of activity. In situations where the analytical results of a group of samples are near the minimum detectable concentration, the 95% confidence interval for the mean may not include zero and thus appear to be statistically significant even though no individual sample contained detectable radioactivity.

Unpaired t-tests were used to determine whether the annual means for the boundary stations were greater than the annual means for the distant stations. All statistical tests used a level of significance of 5% ($\alpha = 0.05$).[C-1]

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