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SURPLUS FACILITIES
MANAGEMENT PROGRAM

POST REMEDIAL ACTION SURVEY REPORT FOR THE SODIUM REACTOR EXPERIMENT (SRE) FACILITY SANTA SUSANA FIELD LABORATORIES ROCKWELL INTERNATIONAL VENTURA COUNTY, CALIFORNIA

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DISCLAIMER

PREFACE AND EXECUTIVE SUMMARY

The Sodium Reactor Experiment (SRE) Facility at the Santa Susana Field Laboratories in Ventury County, California, was built between 1955 and 1957. The 20-MWt sodium-cooled, graphite-moderated, thermal reactor was operated from April 1957 through February 15, 1964, by Rockwell International's Energy Systems Group. During the operational life of the reactor, induced radioactivity was produced by neutron activation of surrounding material. In addition, mixed fission products were released into the reactor primary sodium coolant system when a fuel element ruptured. From 1964 to 1976, the reactor was in a protective storage mode with routine surveillance.

Decontamination of the SRE began in 1976 and was completed in 1982 by Rockwell personnel. At the request of the U.S. Department of Energy (DOE) Office of Operational Safety (OOS) and in accordance with their programmatic concurrent and post-remedial-action responsibilities, the Argonne National Laboratory (ANL) Radiological Survey Group (RSG) conducted a series of radiological measurements and analyses at SRE between November 1979 and September 1982, to survey for residual radioactive contamination.

The survey activities consisted of conducting instrument and smear surveys for radioactive contamination and collecting and analyzing material samples. A total of 80 soil, concrete, rock, rebar (½-in-diameter steel reinforcing bars in concrete), water, and sewer sludge samples were collected by ANL and Rockwell personnel and analyzed at the ANL Analytical Chemistry Laboratory. The samples were collected during various stages of the decontamination and overview activities. The soil, concrete, asphalt, sand, and rebar samples that had elevated levels of radionuclides were those taken from areas that also yielded elevated instrument readings before the decontamination activities had been completed. These areas were subsequently decontaminated to the point that radiation levels were reduced to levels below detectable limits.

During the final overview surveys in the summer of 1982, 46 locations with elevated activity were detected with survey meters. Of these areas, 27 had contamination in excess of acceptable limits; however, all the areas subsequently were cleaned to below detectable limits before ANL/RSG completed their overview activities.

Sludge samples were taken from five access points in the sanitary sewer and storm sewer systems that service SRE. Elevated concentrations of the radionuclide ^{137}Cs were detected in the three samples from the storm sewer system, and ^{90}Sr, ^{137}Cs, and uranium were detected in the samples from the sanitary sewer system as it leaves SRE. The concentration of ^{90}Sr dissolved in the water of one sample was approximately half the recommended water concentration for controlled areas and approximately 15 times the recommended water concentration for uncontrolled areas as stated in DOE 5480.1 Chg. 6, Chapter XI. Past experience has shown that radionuclide concentrations of samples retrieved from sewer access points tend to be considerably lower than concentrations found in straight runs. It must, therefore, be concluded that the sewer systems are indeed contaminated. Past experience has also shown that use of powered plumber's snakes, high-volume water flushing, and detergents are ineffective decontamination procedures; therefore, these options cannot be considered adequate in this instance.

The effluent from the outfall of these drain systems must also be considered potentially contaminated, since experience has shown that radionuclides from contaminated sewers discharge intermittently for many years.

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SANTA SUSANA FIELD LABORATORIES, ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

INTRODUCTION

The Sodium Reactor Experiment (SRE) Facility was built in the mid-1950s at Rockwell International's Santa Susana Field Laboratories (SSFL) as part of the United States' nuclear energy research and development effort. The location is in the Santa Susana Mountains in Ventura County, California, northwest of the San Fernando Valley and 29 miles northwest of downtown Los Angeles (Fig. 1).

The facility, owned by the Federal Government, was constructed on land leased with option to buy from Rockwell International. It was operated by the Rockwell Energy Systems Group (ESG) under a Government contract. The SRE was a 20-MWt, sodium-cooled, graphite-moderated, thermal reactor.[1] Initial operation began in April 1957. The reactor was used for numerous testing programs, and beginning in November 1957, it also was used to generate a supply of electricity sufficient to light the City of Moorpark, California. Operation of the SRE as a nuclear plant ceased after February 15, 1964. During its operating life, the SRE generated more than 37 million kilowatt hours of electrical energy in over 27,300 reactor operating hours.

When the intended program was completed, the SRE was placed in a protective storage mode with routine surveillance. During the operational life of the reactor, induced radioactivity was produced by neutron activation of the surrounding material. In addition, there was a release of mixed fission products into the reactor primary sodium coolant system caused by a ruptured fuel element. In 1974, a plan was formulated with the U.S. Atomic Energy Commission to return the facility to Rockwell International for unrestricted use. Decontamination of the SRE began in 1976 and was completed in 1982 by Rockwell personnel. The decommissioning program included removal of the reactor and surrounding soil and concrete. Current plans are for the SRE area to be used as a non-nuclear fabrication facility.

At the request of the U.S. Department of Energy (DOE) Office of Operational Safety (OOS) and in accordance with their programmatic concurrent and post-remedial-action responsibilities, the Argonne National Laboratory (ANL) Radio-

logical Survey Group (RSG) conducted a series of radiological measurements and analyses at the SRE Facility for the purpose of post-remedial-action assessment. During May 1981, a confirmatory overview survey was conducted at the large area that had been excavated to remove the reactor and the surrounding concrete and soil (see Fig. 2). An onsite radiological survey was performed, and soil, water, concrete, rock, and rebar (½-in-diameter steel reinforcing bars in concrete) samples were collected prior to backfilling operations. A concurrent remedial-action-overview survey was conducted in October 1981 while the backfilling operation was in progress. A post-remedial-action (certification) survey was initiated in July 1982. During August 1982, decontamination and decommissioning (D&D) activities continued at the SRE. Contaminated areas identified by the ANL survey team were also decontaminated. The post-remedial-action survey was then completed during September 1982.

The portion of the SSFL operated by ESG under a long-term lease/option to the U.S. Government is shown in Figure 3. SRE personnel divided the site into ten regions on the basis of topographic factors and operations performed in each region (see Fig. 4). Region IV was an auxiliary parking area and a radioactive materials storage facility. Region V originally contained a hot waste storage area and the interim radioactive waste vault for buried gas and liquid holdup tanks. Region VI consisted of rough, inaccessible terrain with sandstone formations several hundred feet high. Region VII (see Fig. 5) was a low area containing a pond retained by an earthen dam. Region I was used for decontaminating equipment items from the SRE. Regions II, III, VIII, IX, and X were predominantly asphalt parking areas adjacent to the three main buildings involved in the decommissioning project. These were Building 163 (120.7 ft x 40 ft), Building 041, (140 ft x 28 ft), and Building 143 (approximately 16,500 ft^2). Regions X and II are shown in Figure 6, which includes the east end of Building 163. Building 143 housed the SRE reactor and formed an eleventh region. Photographs of Regions III, VII, and SRE are shown in Figure 7.

The eastern part of Building 163 was used as a box fabrication shop The southwestern end was used as a cleaning facility for radioactively contaminated materials. The southern section of Building 041 was used as a storage facility for packaged radioactive material. Building 143 housed the reactor and its control rooms and offices. The reactor was located below ground level in the northeastern section of the building. The High Bay area (see Fig. 8), located above the reactor, measured 50 ft from the floor to the ceiling. Adjacent to

the southern end of the High Bay area was a room that was 25 ft in height called the Low Bay area (see Fig. 9).

Building 133 in Region IV was moved from Region I. It was formerly Building 724, a hot oil sodium cleaning facility. Building 133 contained radioactive waste material and was not part of this decommissioning project.

RADIOLOGICAL SURVEY PROCEDURES

General

The radiological survey conducted by ANL/RSG was performed during five visits to the site over three years. During the first visit (November 1979), environmental corings were taken in three locations (see Fig. 4). These soil samples (10-S12 through 10-S14) were used to assess the background concentrations of radionuclides in the soil. During the second visit (May 1981), the excavated area under the High Bay area (see Fig. 10) was surveyed prior to backfilling operations. During this visit, soil, concrete, rock, and rebar samples were collected by ANL personnel from contaminated areas of the SRE excavation. Regions IV, V, VI, VII, and I also were surveyed during the second visit, and the soil and water samples from Region VII were collected. Three additional samples, 10-R57 through 10-S59, were sent by Rockwell personnel to ANL for analysis after the second visit.

During subsequent decontamination work, radioactivity was found to have migrated along a crack under column 11-D on the east side of the High Bay area. Soil samples 10-S92 through 10-S102 were collected by Rockwell personnel after they had cleaned various portions of this crack. These samples were sent to ANL for analysis.

During the third visit to the site (October 1981), the soil used in the backfilling operation was surveyed, as were areas previously found with elevated activity. Soil samples 10-S103 through 10-S108 from the soil that was used as backfill in the High Bay area were subsequently sent to ANL by Rockwell personnel. A holdup tank south of the hot cell was also removed after the visit. This tank had been used to collect and store shower water during the operation of the reactor. Eight soil samples, 10-S109 through 10-S112, were taken from the sides and bottom of the hole after the tank was removed.

During the fourth visit (July 1982), the SRE building was surveyed. The

floors and walls of the High Bay and Low Bay areas were surveyed to the ceiling with the aid of a Condor man-lift (see Fig. 11). In all other areas of the SRE building, floors were surveyed and the walls were surveyed to a height of 2 m (7 ft). A representative selection of accessible overhead structures such as pipes, vents, and light fixtures were surveyed throughout all of the SRE building. Air samples were collected in the SRE during this visit. The floors and roofs of the SRE and paved areas in Regions II, III, IV, V, VIII, IX, and X were surveyed with portable instruments. As seen in Figure 12, the excavated areas had been backfilled to grade level and the parking area was paved before the visits by the survey team in the summer of 1982.

During the final visit (September 1982), the surfaces of floors were surveyed, and walls were surveyed to a height of 2 m (7 ft) in the ancillary buildings 041, 163, and the floor of pad 687 was surveyed (see Fig. 4). Again a representative selection of accessible overhead structures was surveyed in these buildings. Air samples were collected and the ambient radiation levels in the various buildings were measured.

The results of all five survey trips, together with the investigator's conclusions, are included in this report. Three other site locations at SSFL were also investigated by ANL/RSG. The post-remedial-action survey report for the Building 003 Hot Cave Facility is presented in a companion report (ANL-OHS/HP-83-109). The report for the Kinetic Experiment Water Boiler Facility (KEWB) is presented in ANL-OHS/HP-83-110. The report for the Systems for Nuclear Auxiliary Power-8 (SNAP-8) will be presented in a fourth report (ANL-OHS/HP-84-102).

Instrumentation

Five types of portable survey instruments were used to conduct the direct radiological surveys. Two sizes of gas-flow proportional detectors were used with Eberline FAC-4G-3 electronics to monitor for alpha and/or beta-gamma radiation. The larger floor monitor probe with a 325-cm^2 detection area (AC-22) was used to survey the floors, roofs, and paved parking areas and roads. The smaller probe with a 51-cm^2 detection area (AC-21) was used to survey the walls, overhead, and any areas inaccessible with the floor monitor. NaI crystal detectors, 2 in diameter by 2 mm thick (Eberline PG-2 detector with Eberline PRM-5-3 electronics), were used to monitor for low-energy x and gamma radiation as close

to the surfaces as possible (usually 1 cm from the surface). NaI crystal detectors, 1 in diameter by 1 in thick (Eberline PRM-7 μR meter), calibrated with a ^{226}Ra standard source, were used to measure the ambient external penetrating radiation field (in units of μR/h) approximately 1 m (3 ft) from the walls and floors. All accessible areas, both indoors and out, were surveyed with the above instruments. An end-window Geiger-Mueller (GM) detector (Eberline HP-190 with a 7 mg/cm^2 window and Eberline 530 electronics), calibrated with a ^{226}Ra standard source, was used to measure the contact exposure rate (mR/h) of the contaminated areas. All instruments are described in more detail in Appendix 1.

Although ^{239}Pu and ^{90}Sr-^{90}Y standards were used to calibrate the gas-flow instruments, it should be noted that the numerous isotopes that could be encountered exhibit emission energies differing from those of the standards used in the calibrations. Thus, when detecting known isotopes that emit beta energies differing from those of the standards (such as U), a conversion factor for those particular radionuclides was used to determine the appropriate yield. The methods used to determine the conversion factors are described in Appendix 2. All readings of disintegrations per minute per 100 cm^2 (dis/min-100 cm^2), as reported in Table 1, are equated to ^{90}Sr-^{90}Y unless stated otherwise. Since instrument calibrations were to infinitely-thin, flat-plate standards, all reported readings should be regarded as minimal values; no corrections were made for absorption by surface media.

When possible, the isotopes of contamination were identified by performing a gamma-spectral analysis (using an ND-6 portable spectrometer described in Appendix 1) on the contaminated area or item.

Smear Survey

Dry smears were taken at selected locations throughout the three buildings surveyed. Smears were taken on original structures and components such as walls, floors, pipes, and vents. All standard smears were taken with Whatman No. 1 filter paper, 4.25 cm in diameter. A standard smear is accomplished by applying moderate pressure with the tips of the first two fingers to the back of the filter paper and then rubbing the paper over the surface. Smears of ~ 1000 cm^2 (1 ft^2) were normally taken. A smear of 100 cm^2 was taken from any area or object that had an instrument reading higher than "normal" background, or if there was excessive dirt or dust in an area.

Three different instruments were used to measure (count) the contamination on the smears. They were first counted in groups of ten using a 10-wire, flat-plate, gas-flow, proportional detector developed at ANL. This instrument detects alpha and beta particles and x- and gamma-rays. Additionally, two smears of each group were removed and counted in the sensitive Nuclear Measurements Corporation 2π Internal Gas-Flow Proportional Counters (PC-3A, PC-5, also referred to as PC counters), using an aluminized Mylar window (Mylar spun top) over the smear paper. Smears were counted in each detector for both alpha and beta-gamma activity. These instruments are described in detail in Appendix 1.

Air Samples

Air samples ("grab" samples) were collected with a commercial vacuum cleaner modified at ANL for use as a particulate air-sampling device. A flow rate of 40 cubic meters per hour (m^3/h) was used. A 10% portion (5 cm in diameter) was removed from the filter medium after collection and counted for both alpha and beta-gamma activity in the PC counter. The counting results were used to determine the concentrations of radon (^{222}Rn), thoron (^{220}Rn), radon progeny Working Levels (WL), and the concentration of any long-lived radionuclides. Information and assumptions used to determine the radon concentrations are presented in Appendix 3. Separate air samples were collected with a positive displacement pump to check for the presence of actinon.

Ambient External Radiation

A Reuter-Stokes, RSS-111, pressurized ion chamber was used at selected locations to monitor the ambient external penetrating radiation level at 1 m (3 ft).

Sample Collection

As indicated above, samples were collected during all phases of the overview of D&D operations and during the post-remedial-action (certification) surveys. Three 1-ft deep environmental soil core samples were collected from three locations (10-S12 through 10-S14, see Fig. 4) in November 1979, before the excavation of Building 143. At each location a golf-green hole-cutter was used

to obtain the sample. The sample was taken in three 2-in segments labeled A, B, and C. A final segment from 6 in to 1 ft was labeled D (see Fig. 20).

Thirty-one samples were collected in May 1981, (see Figs. 13 and 14) during the excavation and before the backfill operation. The soil samples were taken to about a 2-in depth with a small hand trowl. Concrete samples were taken with a hammer and chisel (see Fig. 15) by removing part of the contaminated material that had been located while surveying the south wall (see Fig. 16). The rock sample was collected by chipping with a rock hammer (see Fig. 17). The rebar sample was removed by a Rockwell employee with a torch (see Fig. 18). The water samples were collected in 1-ℓ Nalgene bottles. Twenty-nine soil and rock samples from the excavated area and from the backfill soil were collected by Rockwell personnel as requested by ANL Health Physicists and sent to ANL for analysis (see Fig. 19).

An additional eight samples were collected by the ANL survey team during the post-remedial-action surveys in July and September of 1982. The asphalt sample (see Fig. 8) was collected using a hammer and chisel to remove part of the paving material that was contaminated. Sludge samples from the sanitary sewer main were obtained by opening the two access points closest to the SRE building and removing water and sludge with a long-handled scoop and a small bottle. The sludge samples from the storm sewer were made by collecting the moist dirt and rubble below the storm grates with a small hand trowel. The two sand samples (see Fig. 22) were taken from the east heating ventilation and air conditioning unit in the Low Bay area (the left unit in Fig. 9) with a long-handled scoop.

Sample Preparation

The soil samples, the three sludge samples from the storm sewer, and the sample 10-ROCK-57 were prepared for analysis by weighing each sample in its "as collected" state, drying it for approximately 48 hours at 80°C, and then re-weighing the sample to determine dry weight. Each sample was then put into a mill jar (8.7 ℓ), and milled for up to two hours. At no time were any rocks in these samples crushed, ground or pulverized, since this would act to dilute, and hence lower, the reported concentration of deposited radioactive material. After sufficient milling, the sample material was sieved. Each fraction (rocks and dross vs. fines) was bagged and weighed separately (see Fig. 20).

The rock sample 10-ROCK-47 and concrete samples were ground since the radioactivity was impregnated in the material itself. The rebar, asphalt, and sand samples, and the water and sludge samples from the sanitary sewer main, were sent directly to the ANL Analytical Chemistry Laboratory (ACL) for analysis.

Weighed aliquots of the milled fines and ground rock and concrete were loaded into screwtop plastic containers for analysis. Aliquots of 100 g were prepared for gamma-spectral analysis, and 5 g for radiochemical (fluorometric) analysis. Every effort was made during sample preparation to prevent cross-contamination. Soil samples suspected of containing elevated levels of radioactivity were processed in equipment separate from the soil samples considered to contain background levels. All processing equipment was scrubbed and air dried before introduction of the next sample.

The water samples and sludge samples from the sanitary sewer main were separated by ANL-ACL into suspended solids and dissolved solids. Each part of the sample was analyzed separately.

Sample Analyses

All samples, except 10-REBAR-34, were analyzed in the ACL by radiochemical separation of uranium followed by a laser uranium-fluorometric technique to determine total uranium concentration (μg/g). All samples were analyzed by a high-resolution gamma spectrometric technique used to identify and quantify reactor fission and activation products such as cesium (^{137}Cs), cobalt (^{60}Co), and europium (^{152}Eu) and to quantify radium (^{226}Ra) and thorium (^{232}Th) concentrations (pCi/g) found naturally in all soils. A few selected samples also were analyzed by radiochemical separation followed by an alpha spectrometric technique to quantify americium (^{241}Am), plutonium (^{239,240}Pu), and thorium (^{228}Th, ^{230}Th and ^{232}Th) isotopic concentrations (pCi/g). Most of these selected samples and a few others were analyzed by radiochemical separation followed by a gross beta technique to quantify strontium (^{90}Sr). One sample also was sent for mass spectrometric analysis to quantify the atom percent of the uranium isotopes (^{234}U, ^{235}U, ^{238}U) because of a discrepancy between the gamma spectrometric ^{226}Ra result and the total uranium as measured by the uranium fluorometric technique. Soil preparation and analysis techniques are described in more detail in Appendix 4.

SURVEY RESULTS

General

The PAC-4G-3 instrument readings recorded during the surveys have been normalized to units of disintegrations per minute per 100 square centimeters (dis/min-100 cm^2) using the factors derived in Appendix 2. The readings are equated to ^{90}Sr-^{90}Y unless otherwise noted. The PAC-4G-3 readings are reported in net disintegration rates; i.e., the background count rates have been subtracted from the gross count rates prior to conversion to dis/min-100 cm^2. Since all alpha readings were measured at the background level, it was not necessary to correct for alpha contribution to the beta-gamma readings. All smear results are reported as background (BKGD) in Table 1. Since all the smears gave background readings, the actual dis/min-100 cm^2 have not been calculated.

As expected, the background levels varied somewhat. The average background readings for all instruments used in the survey are given in Appendix 1. The fraction of the surface area accessible for survey varied from room to room. Percentages of the total area that were accessible for survey are indicated in Table 1.

Instrument and Smear Survey

During the confirmatory overview survey in May 1981, the direct radiological survey instruments indicated the presence of contaminated locations in the areas excavated to remove the reactor below the High Bay area and in the hot cell east of the High Bay. Those locations were subsequently excavated to a greater depth.

During the post-remedial-action surveys in July and September 1982, 46 locations with elevated levels of radioactivity were identified with the survey meters. Twenty-nine of the contaminated locations were in the Building 143 High Bay and Low Bay areas. These areas are located in Figures 22 through 25. Four other locations were identified in other parts of the SRE (see Fig. 27). The instrument readings are given in Table 1 by location number. Sixteen of these locations exceeded acceptable limits for residual contamination given in Appendix 6. These areas of elevated activity are also listed in Table 2. Gamma

spectral analysis at two of these locations (e.g., 6 in Fig. 23 and 9 in Fig. 27) indicated the presence of ^{137}Cs-^{137m}Ba. In one location (149 in Fig. 27), the contaminant was identified as normal uranium. All smear locations and areas of elevated activity are shown in Figures 22 through 26 for the High Bay and Low Bay areas. Figure 21 is an exploded view of the bay areas. The light fixtures in the High Bay area were known to be contaminated. Since they were still in place at the time of the survey, several of these fixtures were checked for contamination during the overhead survey. These fixtures are included among the 29 contaminated locations in the bay areas. The smear locations for the SRE roofs and the exhaust pipe from the hot cell exhaust equipment room (170) are shown in Figure 28. The smear and contaminated area locations for the remainder of the first floor of SRE are shown in Figure 27. Figure 29 locates the smears taken on the mezzanine (second floor) and in the hot cell (below the High Bay area).

Building 041 had seven areas of elevated activity, with four of these above acceptable limits of contamination (see Fig. 30 and Table 2). Pad 687, behind Building 041 (see Fig. 4) had two areas of activity above acceptable limits (see Fig. 31 and Table 2). Four locations with activity above acceptable limits, two in Region II and two in Region IX, were identified in the paved areas (see Fig. 27 and Table 2). The paving in the northwestern corner of Region IX was removed and the area was in the process of being decontaminated. During the September visit, this area was checked and residual contamination was found as indicated in Table 2. The locations of smears taken in Building 163 are shown in Figure 32.

All 46 locations with elevated levels of radioactivity were cleaned to below detectable limits by Rockwell personnel before termination of the post-remedial-action survey.

Air Samples

A total of nine "grab" air samples were collected during the two visits in 1982. The results of the radon and thoron concentrations and the Working Level (WL) determinations are given in Table 3. The locations where samples were collected are identified in Figures 23, 27, 29, 30, and 32. Five samples were taken with the positive displacement pump as indicated in Table 2. The location of these samples was the same as the corresponding locations of the "grab"

samples. No actinon progeny was detected in any air sample. The radon progeny level, ranging from 0.0005 to 0.0125 WL, was below the 0.02 WL (including background) level stated in 40 CFR Part 192.12 as the objective of remedial action. Radon (^{222}Rn) concentrations, ranging from 0.05 to 1.25 pCi/ℓ, and thoron (^{220}Th), ranging from 0.002 to 0.027 pCi/ℓ, were below the 3 pCi/ℓ and 10 pCi/ℓ concentrations respectively, specified in the DOE 5480.1, Chapter XI "Requirements for Radiation Protection" for uncontrolled areas. There was no long-lived activity above detectable limits. Therefore, the air samples were all within acceptable levels for uncontrolled areas as specified in the regulatory guidelines given in more detail in Appendix 6.

Ambient External Radiation

The ambient external penetrating radiation levels that were measured in five locations, as indicated in Table 4 and Figures 23, 30, and 32 ranged from 9.9 to 13.5 μR/h. The PRM-7 background measurements were approximately 15% below these readings. In both cases, the ambient external gamma radiation levels were below the 20 μR/h above background standard set in 40 CFR 192.12 for occupied buildings (see Appendix 6).

Material Samples

A total of 62 soil samples were collected from the SRE area, either by the ANL survey team or by Rockwell personnel, and sent to ANL for analysis. Three environmental samples, 10-S12 through 10-S14 (each collected to a 1-ft depth and divided into four sections) were taken in November 1979 to determine the local concentrations of radionuclides. The collection points are shown in Figure 4. The sample weights are listed in Table 5. The results from the gamma spectrometric and uranium fluorometric analyses are given in Table 6. These samples contained concentrations of the ^{232}Th decay chain, the ^{226}Ra decay chain and uranium normal for soil from this area of the country. However, the concentration of ^{137}Cs in the soil samples 10-S-12B through 10-S-12D, from 5 cm (2 in) to 30 cm (1 ft) below grade level contained approximately twice the normal concentration. Two of these environmental samples (10-S-12C and 10-S-13D) had low, but detectable, concentrations of ^{60}Co. The maximum ^{137}Cs concentration of these three samples (1.56 pCi/g) was well below the "Interim Soil Limits for D&D

Projects" of 80 pCi/g as recommended in LA-UR-79-1865-Rev. The ^{60}Co concentrations, less than 0.1 pCi/g, were well below the accepted limits for residual radioactivity for beta-gamma emitters of 100 pCi/g even when combined with the ^{137}Cs concentration.

Twenty-two soil samples, five concrete samples, a rock, a rebar, and two water samples (10-S-29A through 10-S-56) were taken by the ANL team in May 1981 during the concurrent-remedial-action survey while SRE was excavated. The sample locations are shown in Figures 13 and 14. The sample weights are listed in Table 5 and the gamma spectrometric analyses for ^{137}Cs, ^{232}Th, and ^{226}Ra, and the uranium fluorometric analysis are presented in Table 6. The samples analyzed for induced ^{90}Sr activity and those found by gamma spectral analysis to have induced ^{137}Cs, ^{60}Co or ^{152}Eu are presented in Table 7. The samples with elevated ^{137}Cs concentrations (10-S-31, 10-S-48, 10-C-37A and B) were taken from areas that gave elevated readings on the portable survey instruments. Three of the samples also had elevated concentrations of ^{60}Co or ^{90}Sr. Suspect samples were also analyzed to determine the plutonium ($^{239,240}Pu$) and americium (^{241}Am) concentrations and the thorium isotopic (^{228}Th, ^{230}Th, and ^{232}Th) concentrations. The plutonium, americium, and thorium were chemically separated and the relevant concentrations were determined by alpha spectrometric analysis. The results presented in Table 8 indicated negligible concentrations of plutonium and americium. The ^{232}Th is in equilibrium with ^{228}Th and in concentrations expected to occur naturally.

Sample 10-S-31 was taken from a pit that contained drain lines. One soil sample, 10-S29-A, had a slightly elevated uranium concentration. This sample was taken from another pit in the SRE building (see Fig. 13). For natural uranium, there should be 5.6 ± 0.5 pCi/g of uranium for 2.8 ± 0.2 pCi/g of ^{226}Ra in 10-S29-A where 8.0 ± 0.8 pCi/g was actually measured. To check the abundances of uranium isotopes, a mass spectrometric analysis was performed to check for possible contamination by depleted or enriched uranium. The results given at the bottom of Table 6 indicate isotope ratios found naturally. To further check for the possible contamination by normal uranium (uranium with isotopic abundances found in nature but separated from its long-lived decay products), three additional samples, 10-R57, 10-S58, and 10-S59, were sent to ANL by Rockwell personnel from the same pit as requested. The locations of all the samples collected by Rockwell personnel (10-ROCK-57 through 10-S-59; 10-S92-24 through 10-S112-8.0 and 10-S130) are shown in Figure 19, and the results of the

analyses are presented in Table 6. All three samples indicated the expected uranium/radium ratio. The concentrations of both radionuclides were slightly lower in these samples than in 10-S29-A. However, they were similar to the results from 10-S29-B. The plutonium concentration was low and thorium isotopic abundances for these samples indicated natural thorium (see Table 8).

During the decontamination work, activity was found to have migrated along a crack (an area of high permeability for radionuclides through the soil) under a column 11-D on the east side of the High Bay area. The contaminated crack continued east of the building, then down to a deeper level, where it doubled back into the High Bay area and then under Column 11-D again and out of the east side of the building. Eleven soil samples, 10-S92 through 10-S102, were collected by Rockwell while the firm was decontaminating this "crack." The results of the analyses indicate that the crack was decontaminated to background concentrations of soil radionuclides. The sample number system consists of four parts: a site designation (10); a sample-type designation (S for soil); a sequential number (92 through 102); and a depth indicator (24). Therefore, the samples from the decontamination of the crack were taken between 24 and 33 ft below grade level. The top of the footing for Column 11-D was 20 ft below grade level.

During the backfilling operations, six soil samples, 10-S103 through 10-S108, were collected from 16.4 ft and 19.7 ft below grade level as requested by ANL-RSG. The concentrations of the radionuclides in all of these samples were at background levels. A small 1000-gallon holdup tank was removed from south of the hot cell. After the tank was removed, eight soil samples were collected for analysis. One sample, 10-S109-4.0, had a slightly elevated ^{137}Cs concentration, 3.09 pCi/g. The rest of the results of the analyses of these soil samples indicated background concentrations of radiounuclides. After the pit that contained the drain lines was finally decontaminated, a final sample, 10-S130, was collected. In May 1981, sample 10-S31 had been collected from the same pit. The gamma spectral results indicated that the ^{137}Cs concentration had been reduced from 680 ± 50 pCi/g to background levels. As indicated in Table 7, the ^{90}Sr concentration was reduced from 4490 ± 1050 pCi/g to 4.8 ± 0.5 pCi/g. Although this concentration is still above background levels, it is below the 100 pCi/g limits for residual radioactivity for beta-gamma emitters that have been accepted for this site.

During the survey visits in July 1982, three samples were collected from contaminated areas. One, 10-ASPHALT-113, was removed from the parking area in Region III (see Fig. 8). The contamination, identified as ^{137}Cs (see Table 6), apparently was from a spill during either the operation or cleanup of the SRE area. The radioactivity was detected with portable survey instruments and is identified as a point of elevated activity (No. 161 in Fig. 27). The contamination near this spot and an adjacent spot (160), indicated that rainwater had probably washed some of the radioactivity into the storm sewer system. This contaminated area was cleaned to below detectable limits as indicated by a survey of the area in September 1982 with portable instruments. The other two samples, 10-SAND-114 and 10-SAND-115, were taken from inside the east heating ventilation and air conditioning unit. The unit was located in the Low Bay area, as seen in Figures 9 and 22. The samples contained elevated levels of ^{137}Cs and ^{60}Co (see Table 7). These levels were considered significant, particularly in an air circulation unit. Therefore, the two air treatment units were removed and disposed of before the termination of the survey in September.

During the final trip in September 1982, five samples were taken from the sanitary sewer main and storm sewer drain access points (see Fig. 8) adjacent to the SRE building. The storm drain samples, 10-SS-125 through 10-SS-127, contained ^{137}Cs contamination, as expected (see Table 7), due to rain water washing the ^{137}Cs contamination located at 10-ASPHALT-113 into the storm drain lines. One sanitary sewer sample, 10-SS-128, contained elevated levels of ^{90}Sr and ^{137}Cs. The ^{90}Sr contamination of 4.7 ± 0.5 pCi/mℓ in the dissolved solids is approximately half the 10 pCi/mℓ soluble water concentration recommended for controlled areas and approximately 15 times the 0.3 pCi/mℓ soluble water concentration recommended for uncontrolled areas as stated in DOE 5480.1 Chg. 6, Chapter XI. The other sanitary sewer sample, 10-SS-129, contained elevated levels of ^{137}Cs and uranium (see Tables 6 and 7). According to the NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestircted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material," dated July 1982, the interior surfaces of drain lines should be assessed for contamination at appropriate access points. Past health physics experience has indicated that the contamination level found at such access points is often lower than that found in straight runs. Therefore, it is assumed that although the ^{137}Cs and uranium contamination found in the samples from the two sewer systems is low, it is a further indication that the sewer lines are

indeed contaminated. The level of contamination is unknown. Past experience has also shown that use of powered plumber's snakes, high-volume water flushing, and detergents are ineffective decontamination procedures. Therefore, these options cannot be assumed to be adequate in this instance. Health Physics experience has also noted contamination in the effluent from similar sewer systems intermittently for many years.

CONCLUSIONS

In view of the concurrent and post-remedial-action surveys, the following conclusions can be stated:

1) All the buildings and areas included in this decommissioning project (those within the heavy black lines in Figure 3, except Building 133) have been decontaminated to below the limits specified in the draft ANSI Standard N13.12 and the NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material," dated July 1982.

2) Radioactive contamination was found in appropriate access points of the sanitary sewer and storm drain systems included within the boundaries of this decommissioning project. One sample indicated a ^{90}Sr concentration dissolved in the water of approximately half the recommended water concentration for controlled areas and approximately 15 times the recommended water concentration for uncontrolled areas as stated in DOE-5480.1 Chg. 6, Chapter XI. Therefore, the interior inaccessible surfaces of these systems must be considered contaminated in accordance with statements found in the NRC Regulatory Guidelines issued in July 1982. Effluent from the outfall of this drain system must also be considered as being potentially contaminated.

REFERENCE

1. U.S. Department of Energy. 1982. "Sodium Reactor Experiment Decommissioning Environmental Evaluation Report." Rockwell International, Energy Systems Group DOE-SF-4 (ESG-DOE-13367).

Los Angles Vicinity

ANL-HP DWG. NO. 80-71

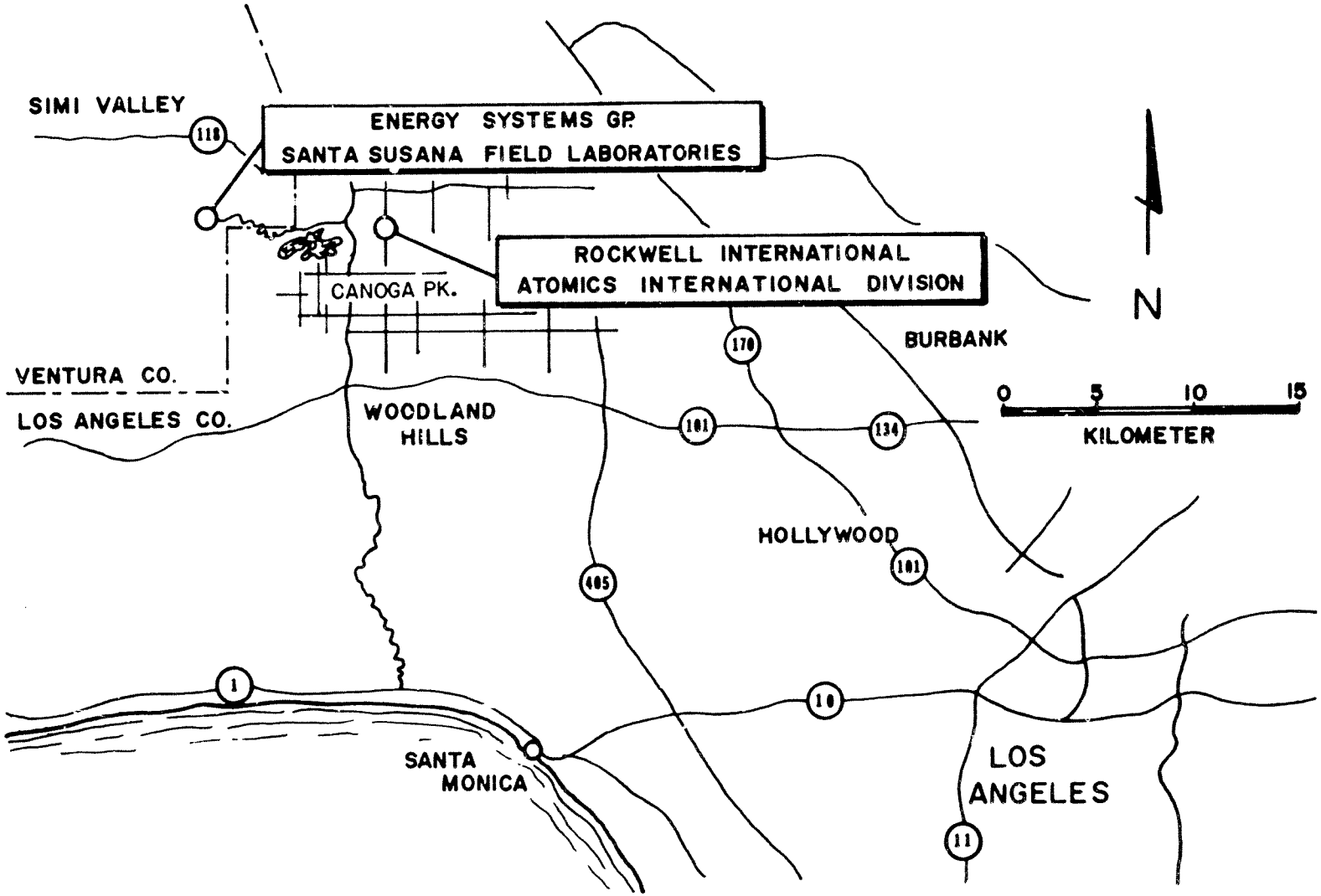


Figure 1

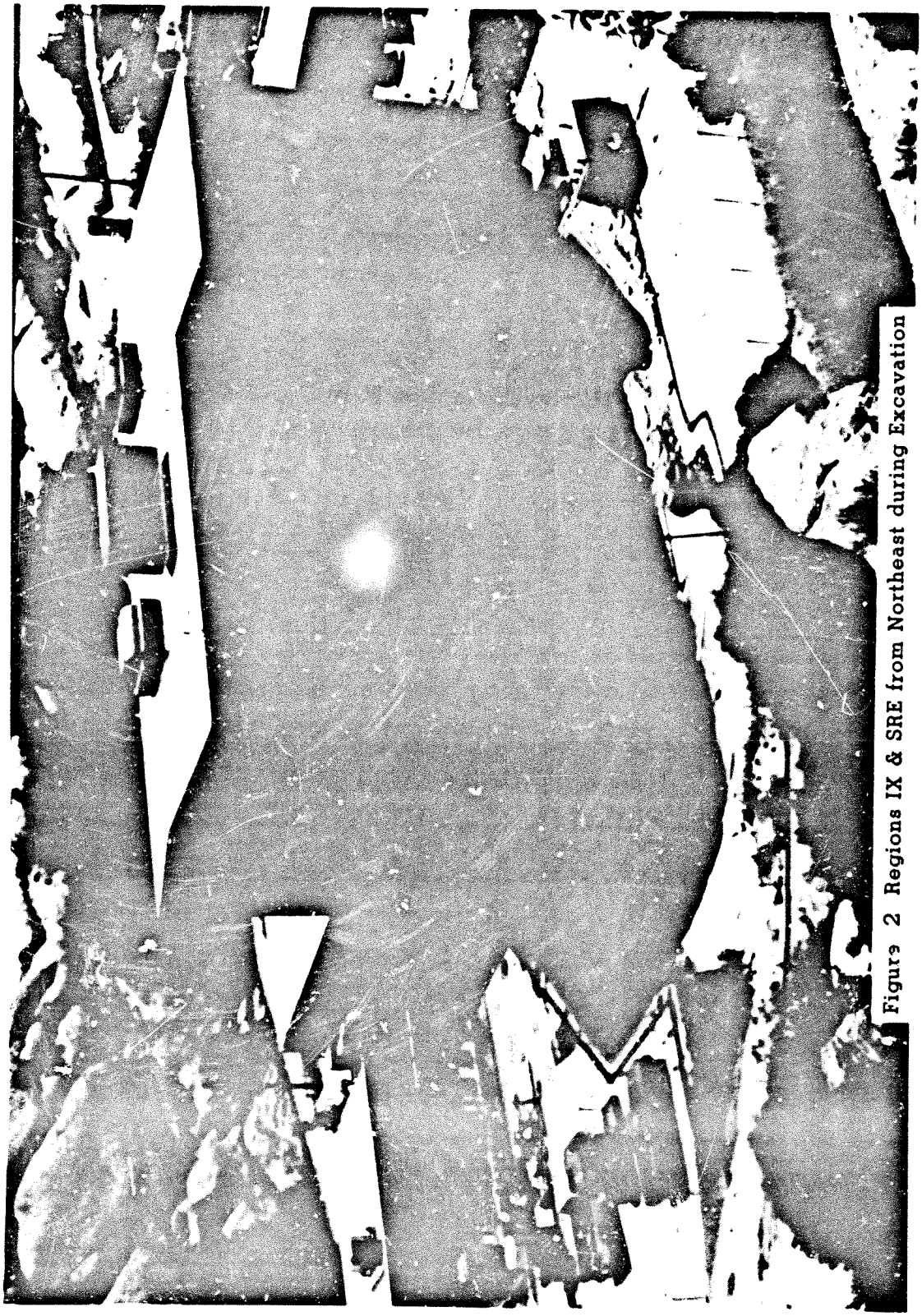
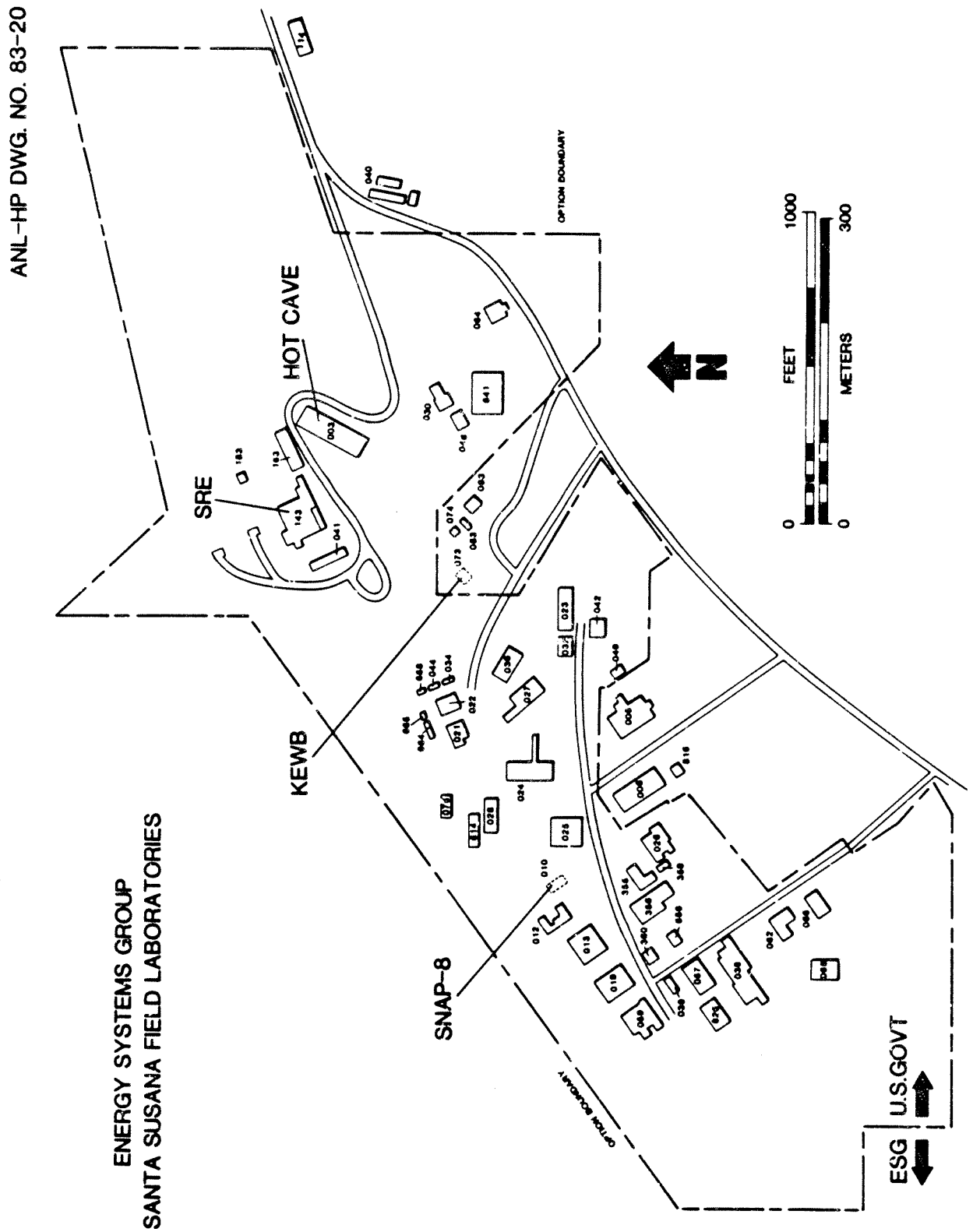


Figure 2 Regions IX & SRE from Northeast during Excavation

Figure 3


ANL-HP DWG. NO. 83-20
ENERGY SYSTEMS GROUP
SANTA SUSANA FIELD LABORATORIES
HOT CAVE
SRE
KEWB
SNAP-8
OPTION BOUNDARY
N
FEET
METERS
1000
300
0
0
ESG
U.S.GOVT

SRE Environs and Environmental Soil Sample Locations

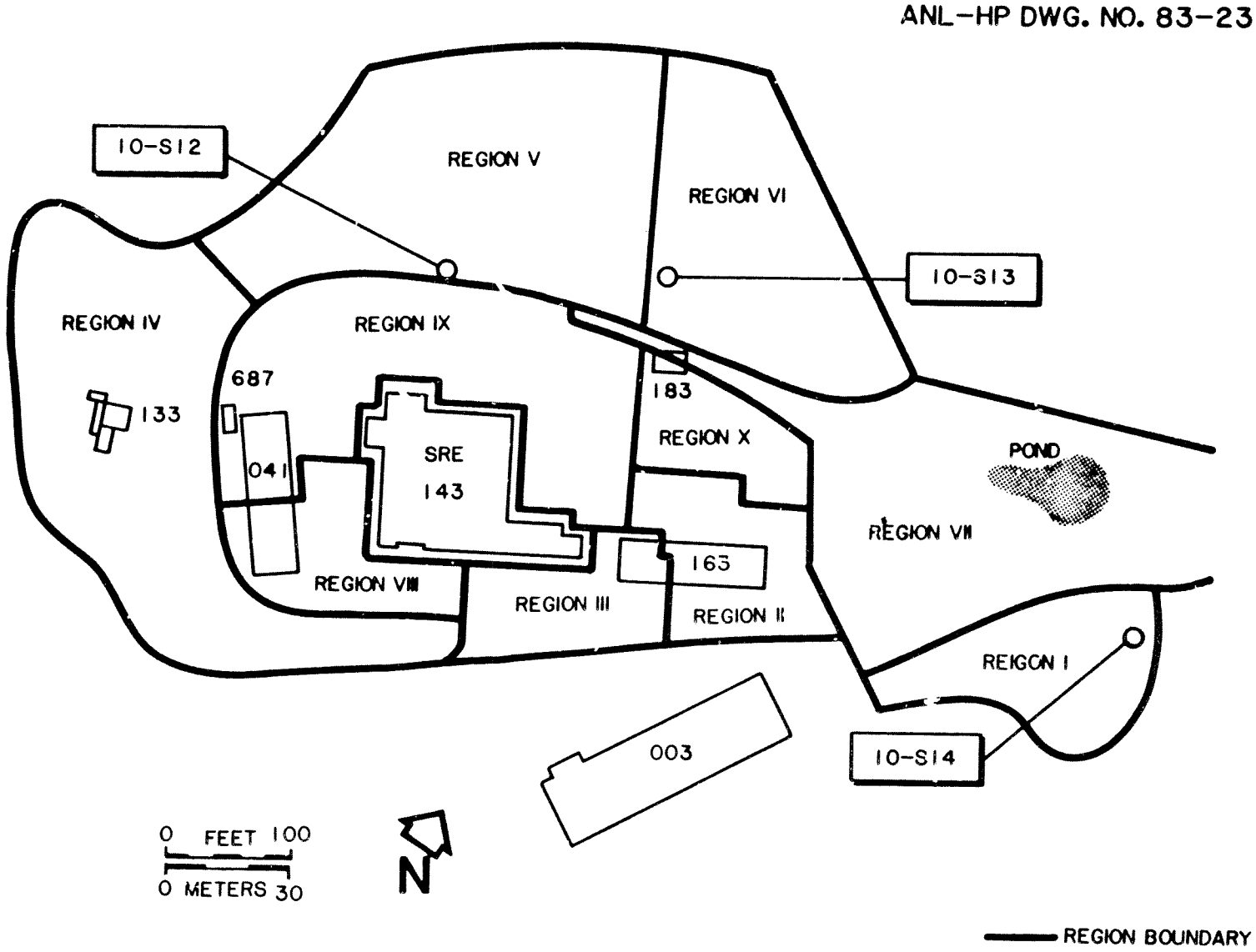


Figure 4

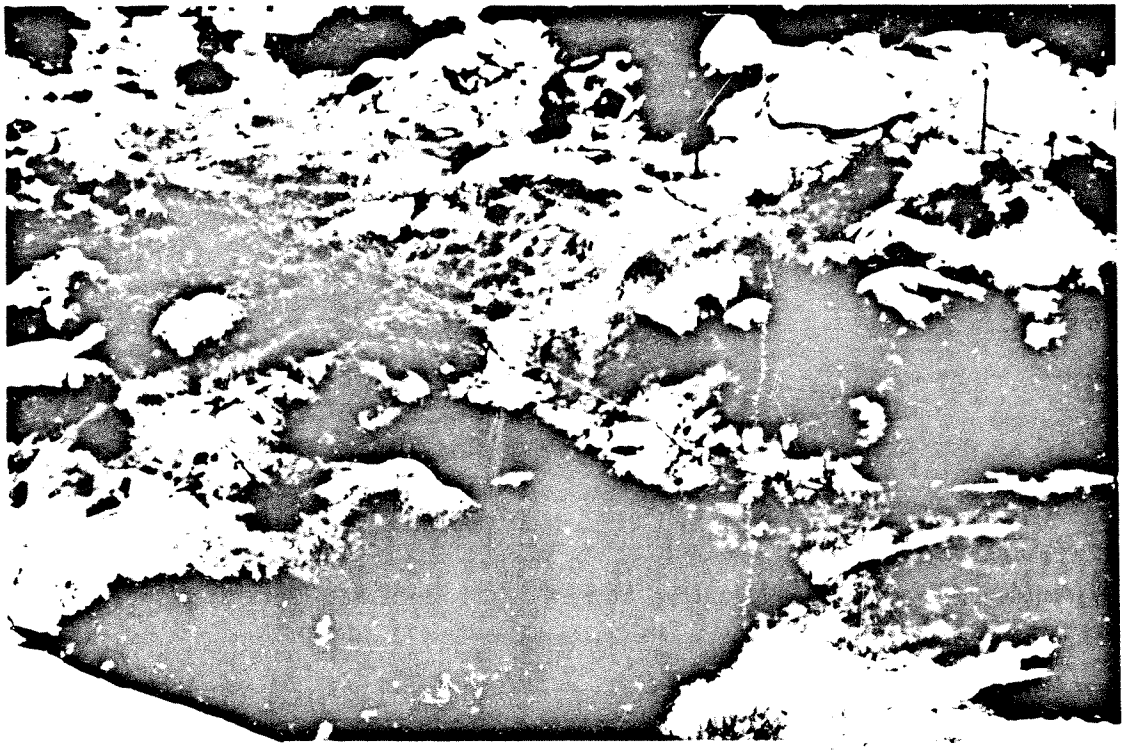


Figure 5 Region VII with Holdup Pond

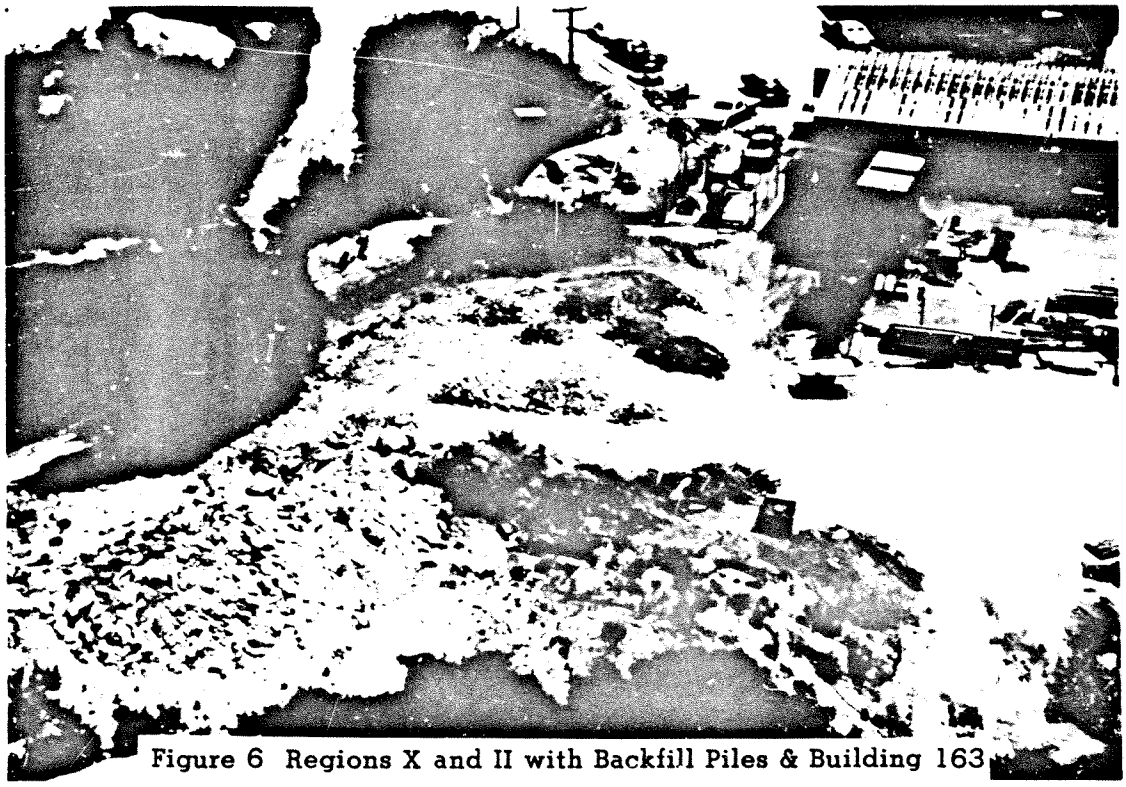


Figure 6 Regions X and II with Backfill Piles & Building 163

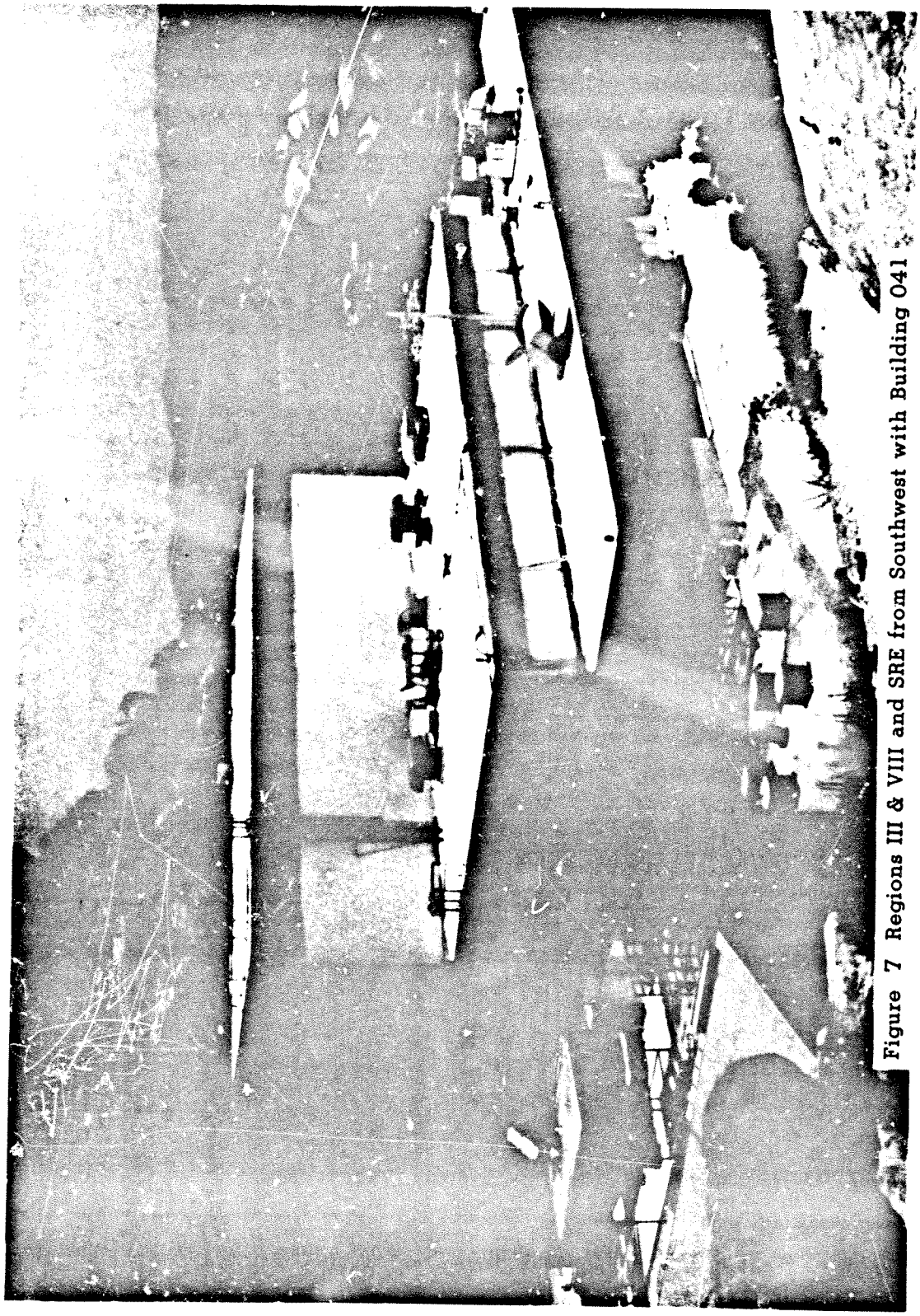


Figure 7 Regions III & VIII and SRE from Southwest with Building 041

Sanitary and Storm Sewer Sample Locations

ANL-HP DWG. NO. 83-24

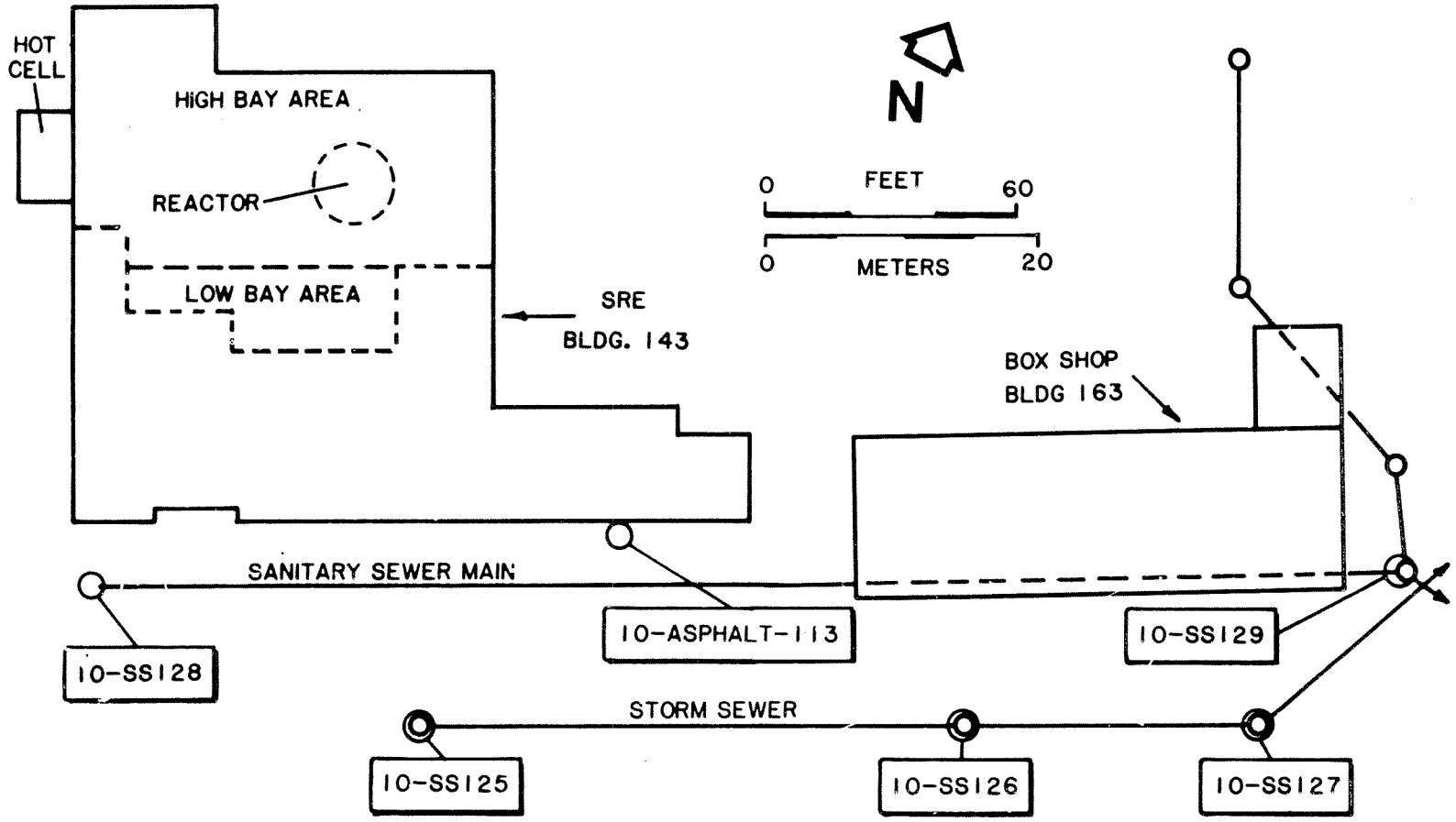


Figure 8

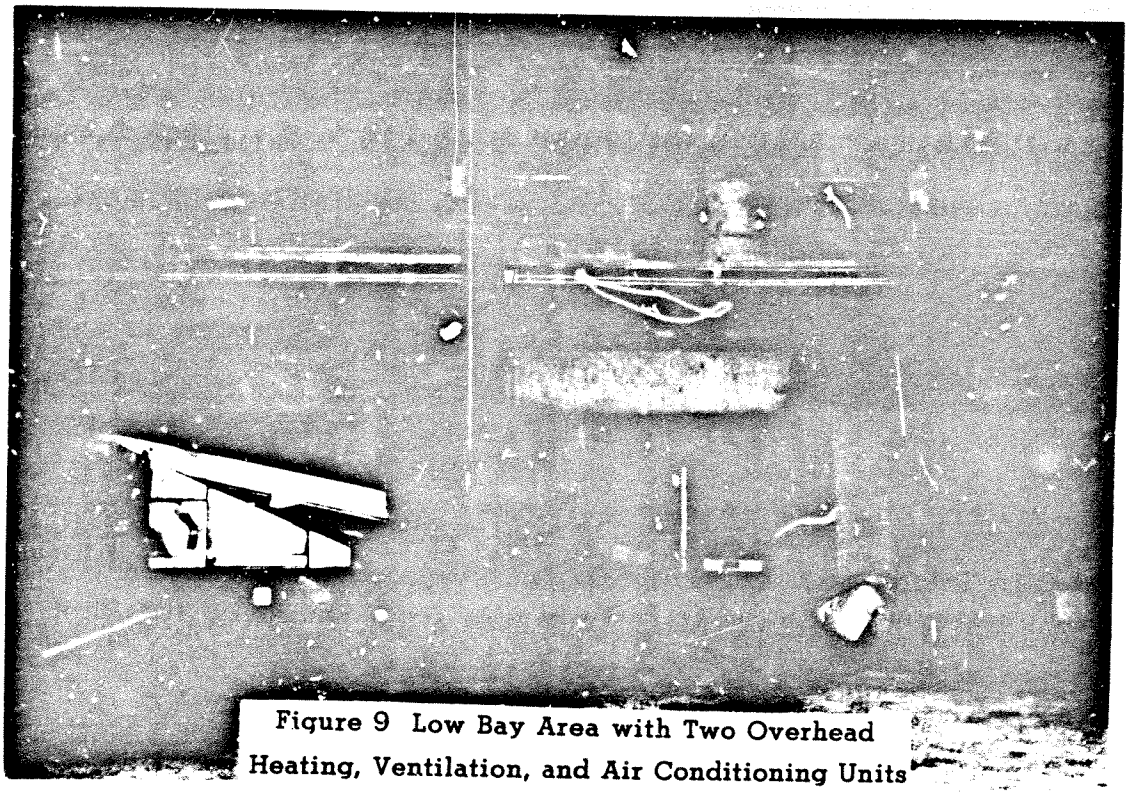


Figure 9 Low Bay Area with Two Overhead Heating, Ventilation, and Air Conditioning Units

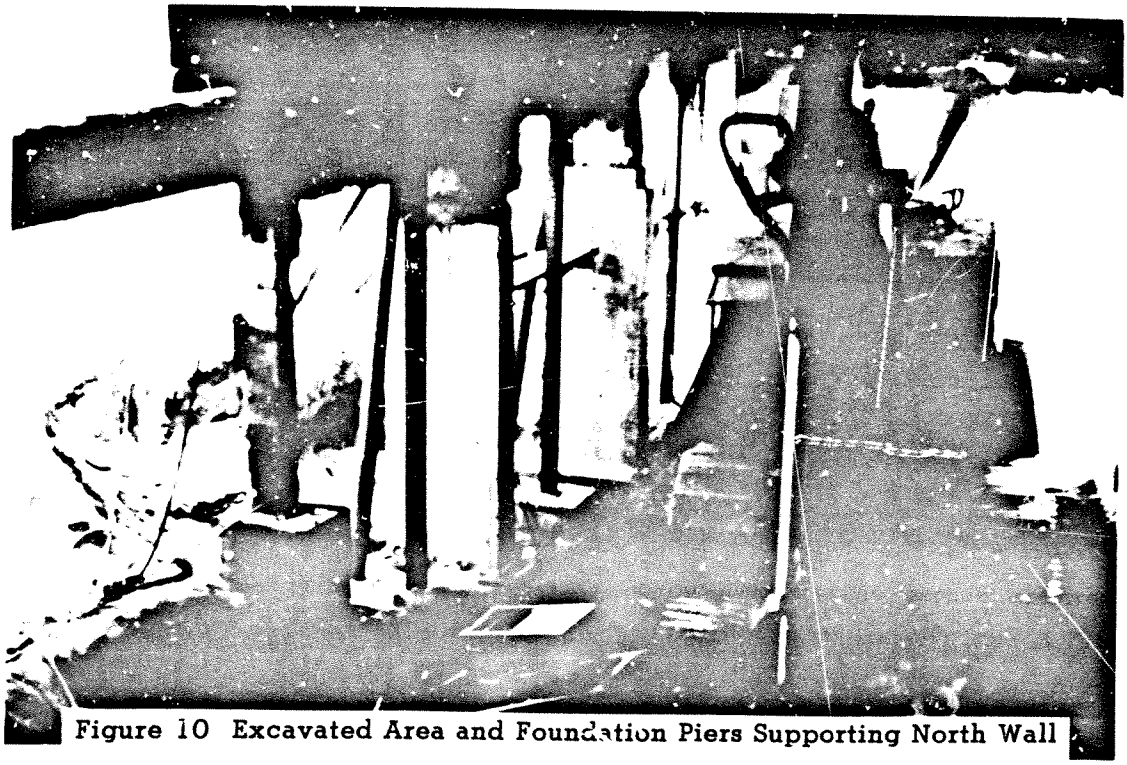


Figure 10 Excavated Area and Foundation Piers Supporting North Wall

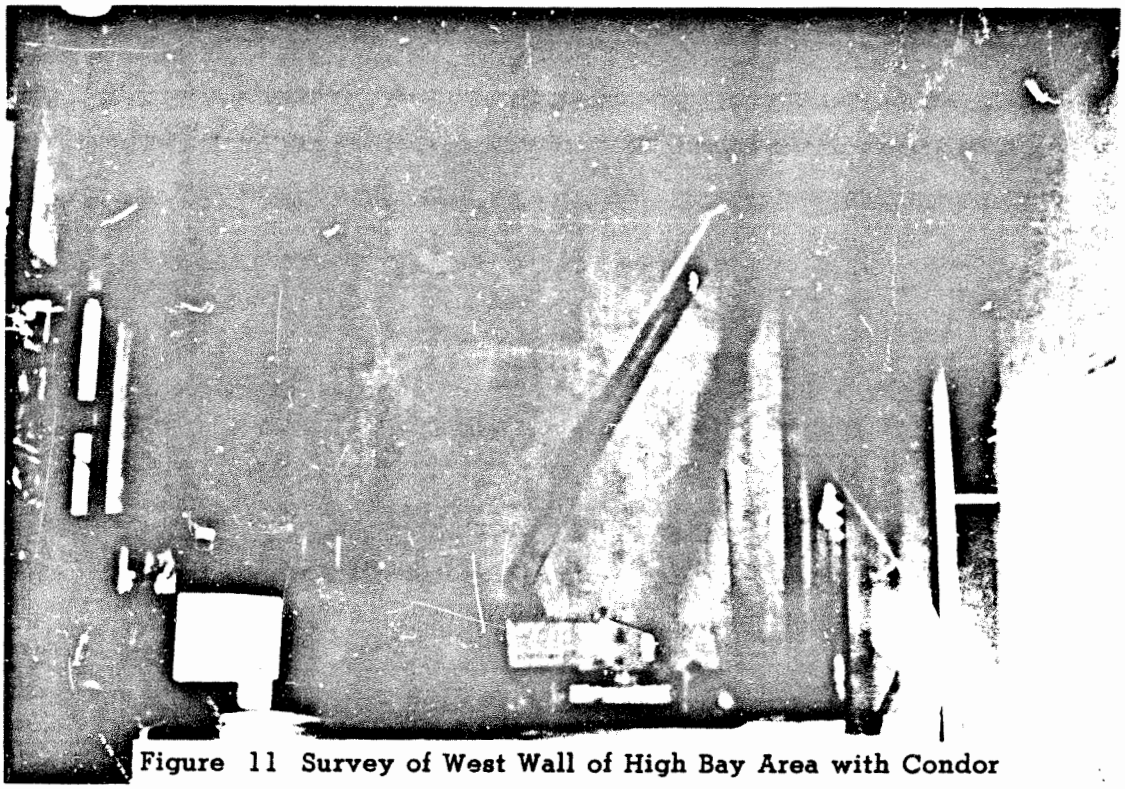


Figure 11 Survey of West Wall of High Bay Area with Condor

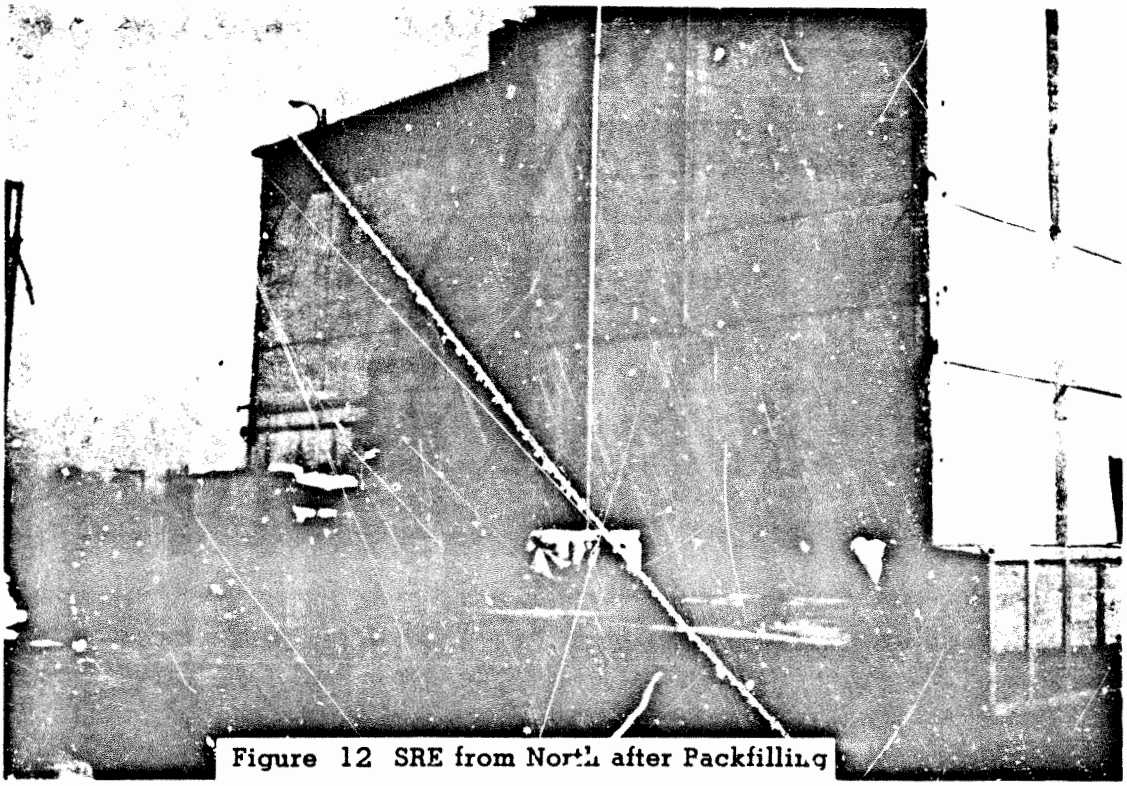


Figure 12 SRE from North after Backfilling

ANL Sample Locations During Excavation

AN' -HP DWG. NO. 83-49

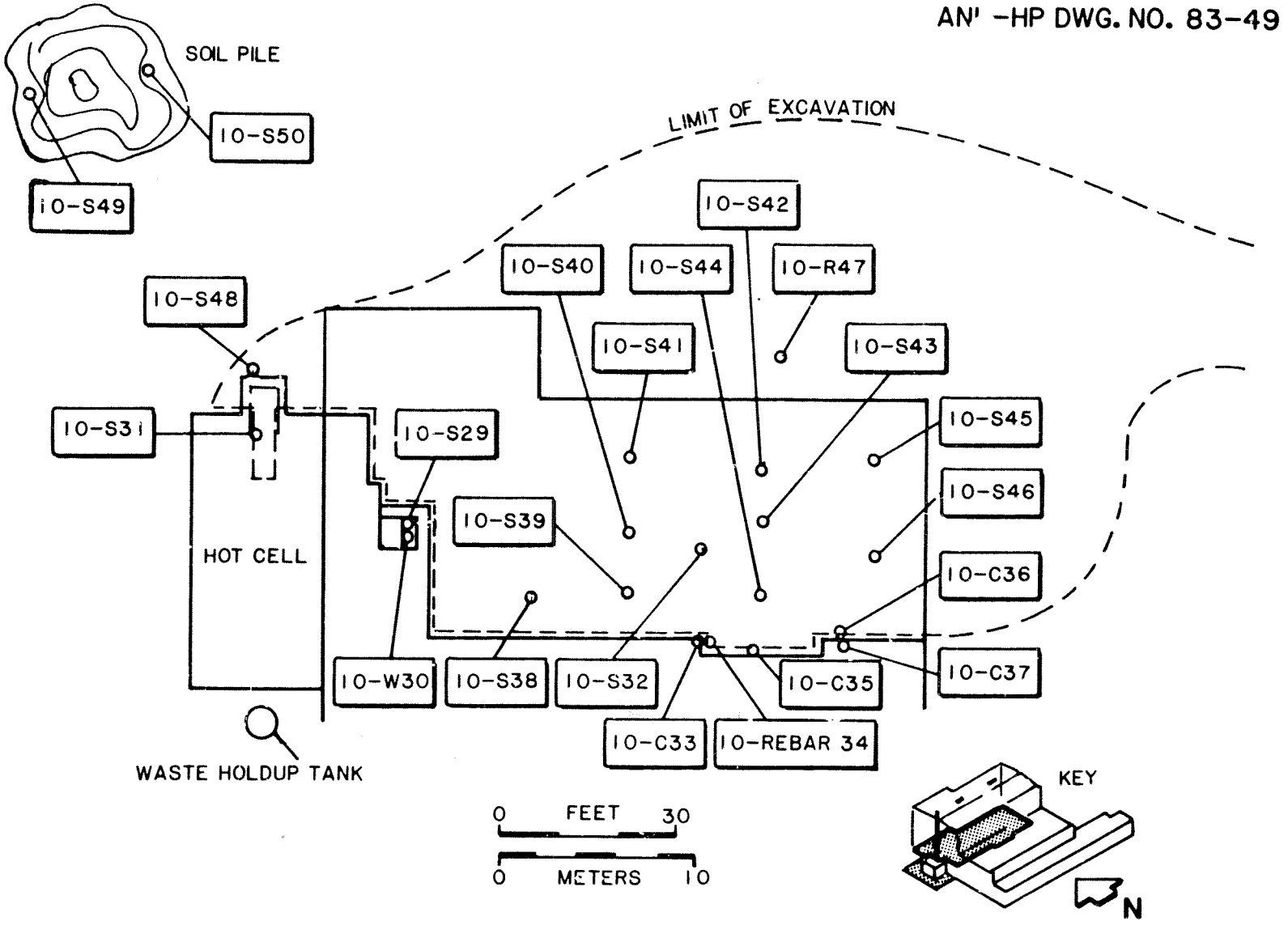


Figure 12

Figure 14

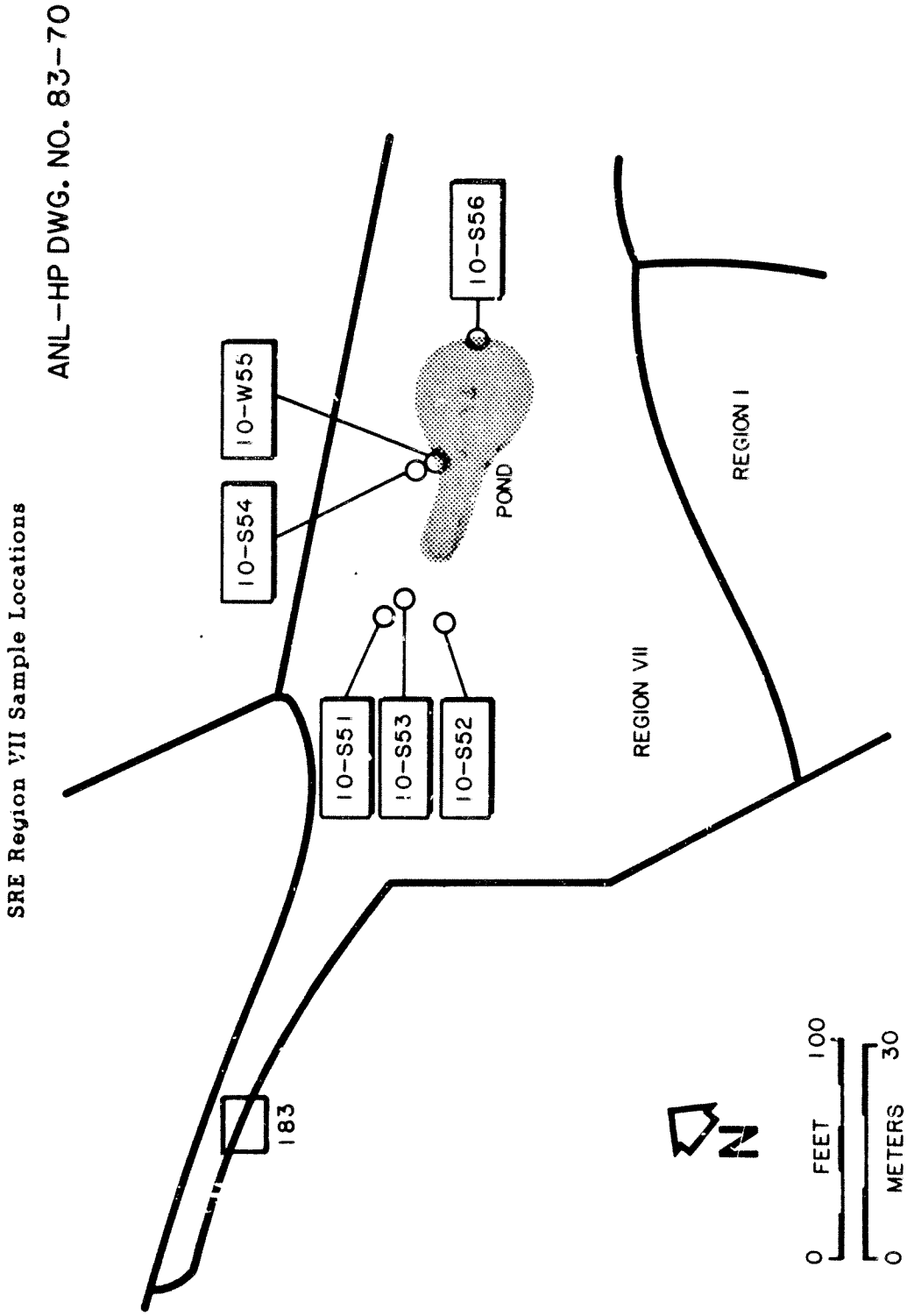

ANL-HP DWG. NO. 83-70
SRE Region VII Sample Locations
10-W55
10-S54
10-S56
POND
REGION I
REGION VII
10-S51
10-S53
10-S52
183
N
0
FEET
100
0
METERS
30

Figure 15
Collection of Concrete Sample #35 from South Wall

Figure 16 South Wall - Surveying from Ladders

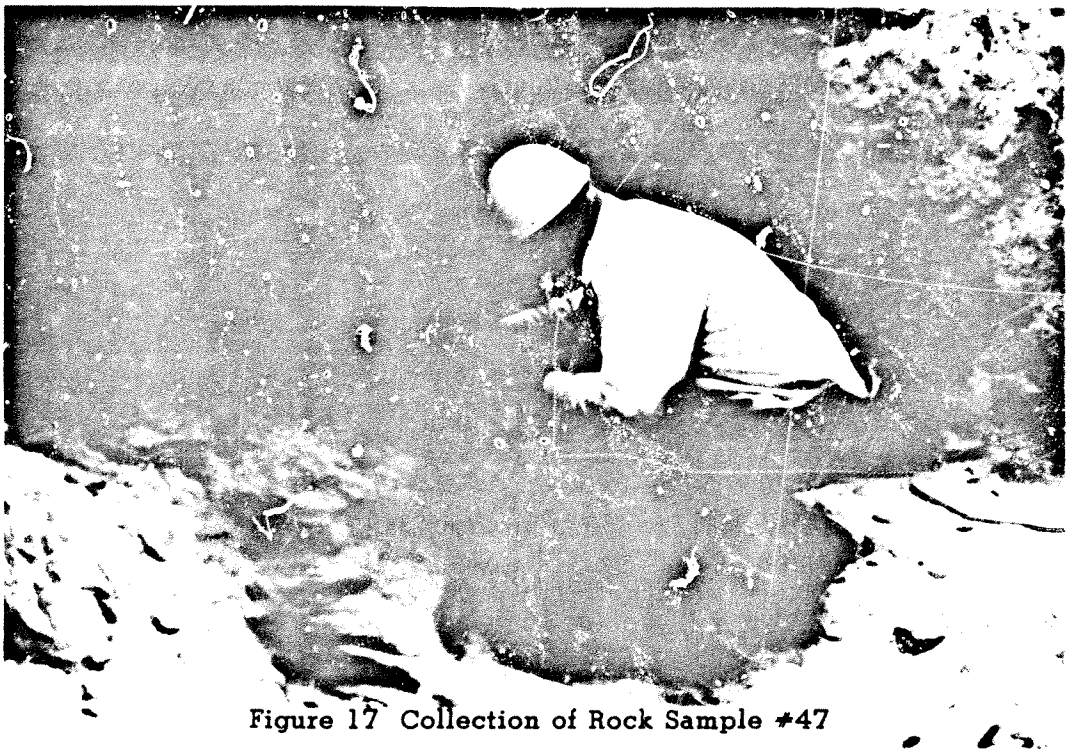


Figure 17 Collection of Rock Sample #47

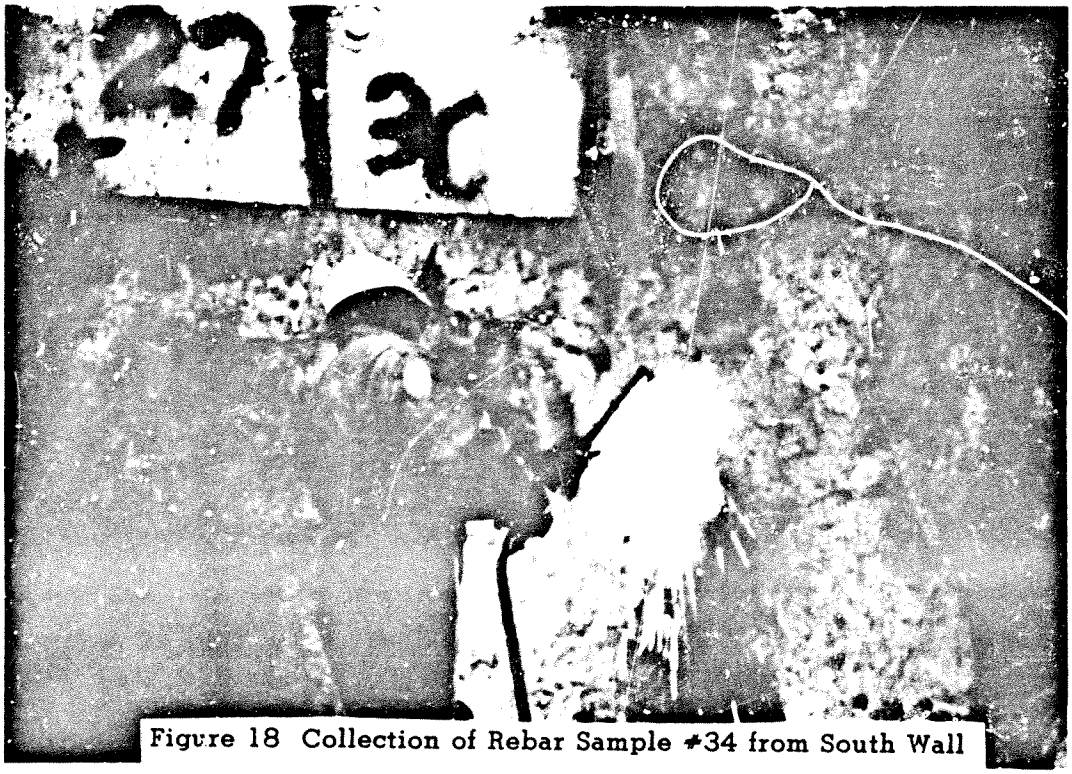


Figure 18 Collection of Rebar Sample #34 from South Wall

Rockwell Soil Sample Locations During Excavation

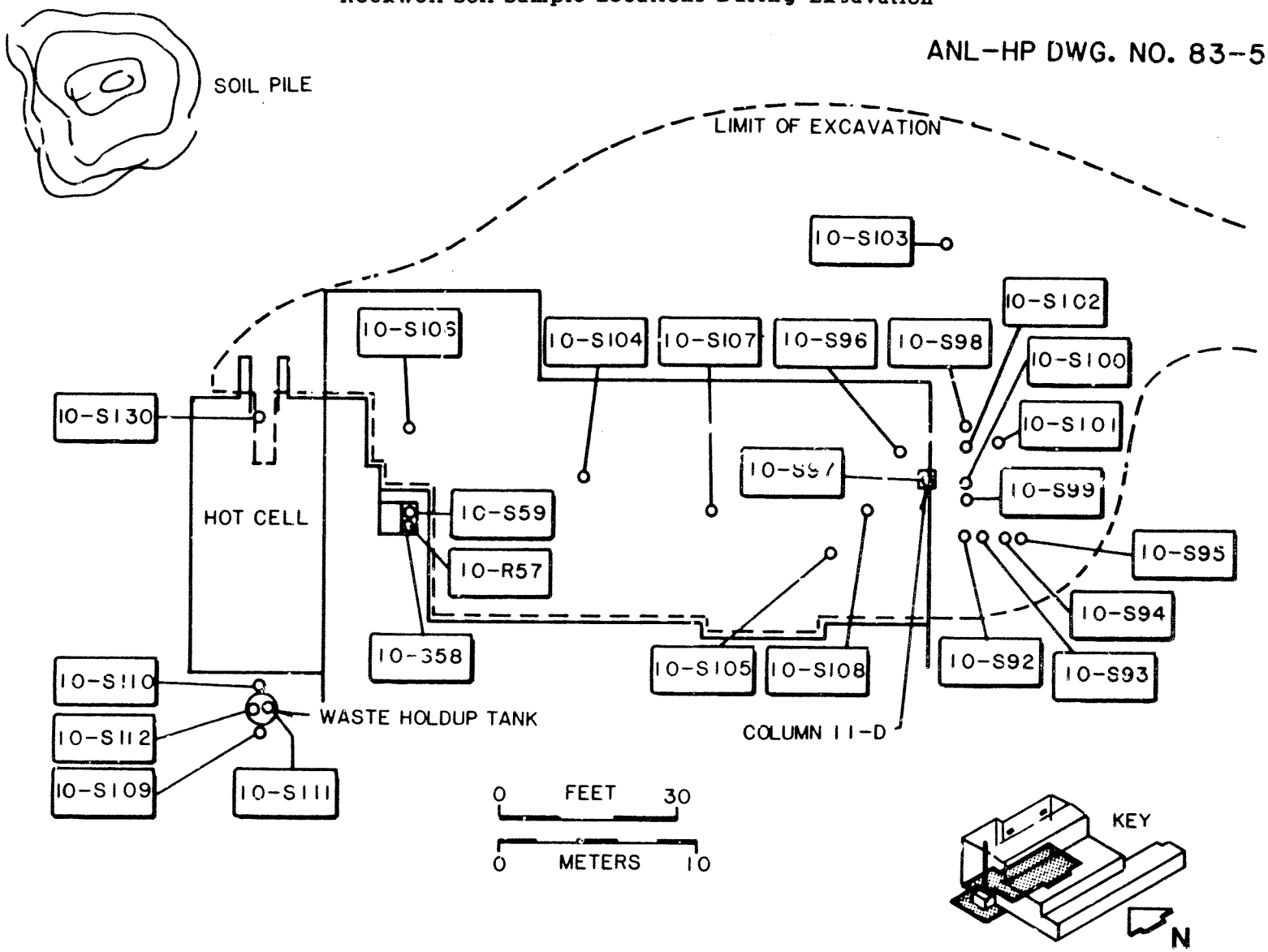
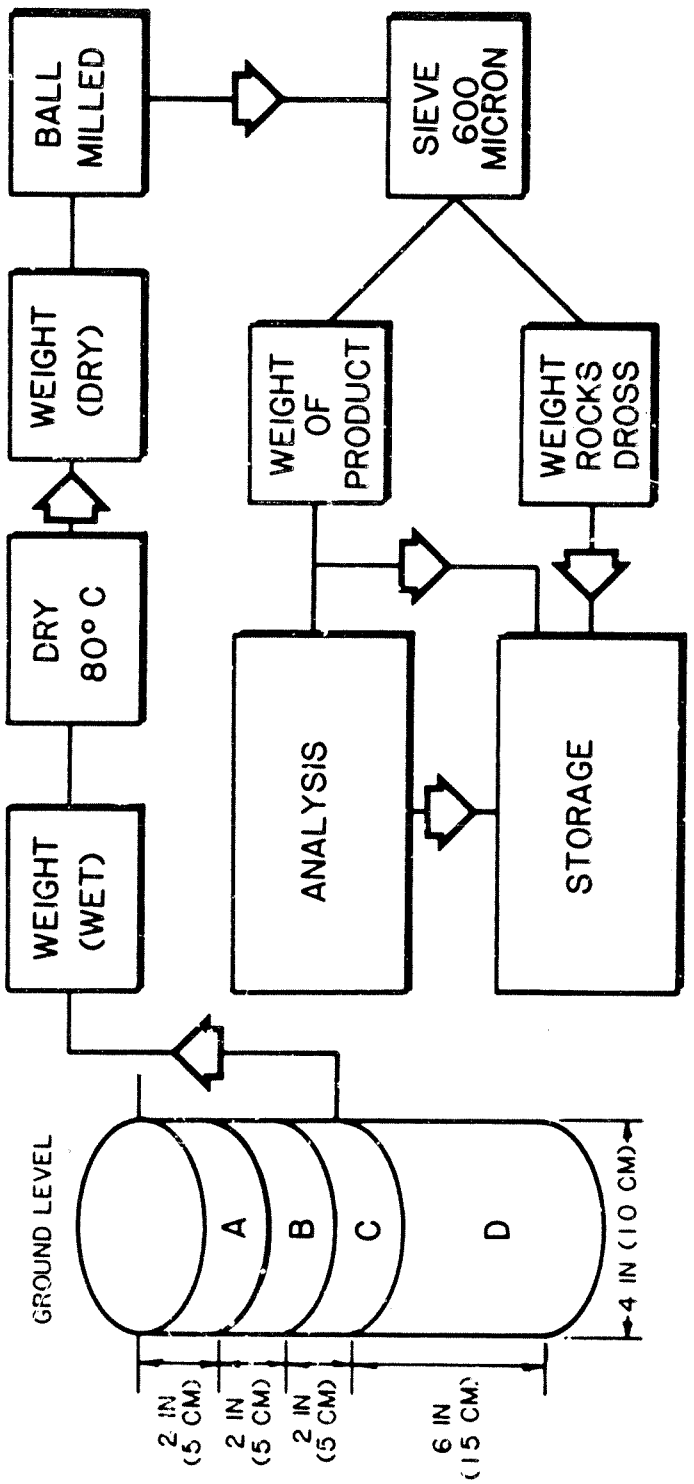


Figure 19

Figure 20

Environmental Soil Sample Processing

ANL-HP DWG. NO. 78-2a



High Bay and Low Bay Surfaces

ANL-HP DWG. NO. 83-52

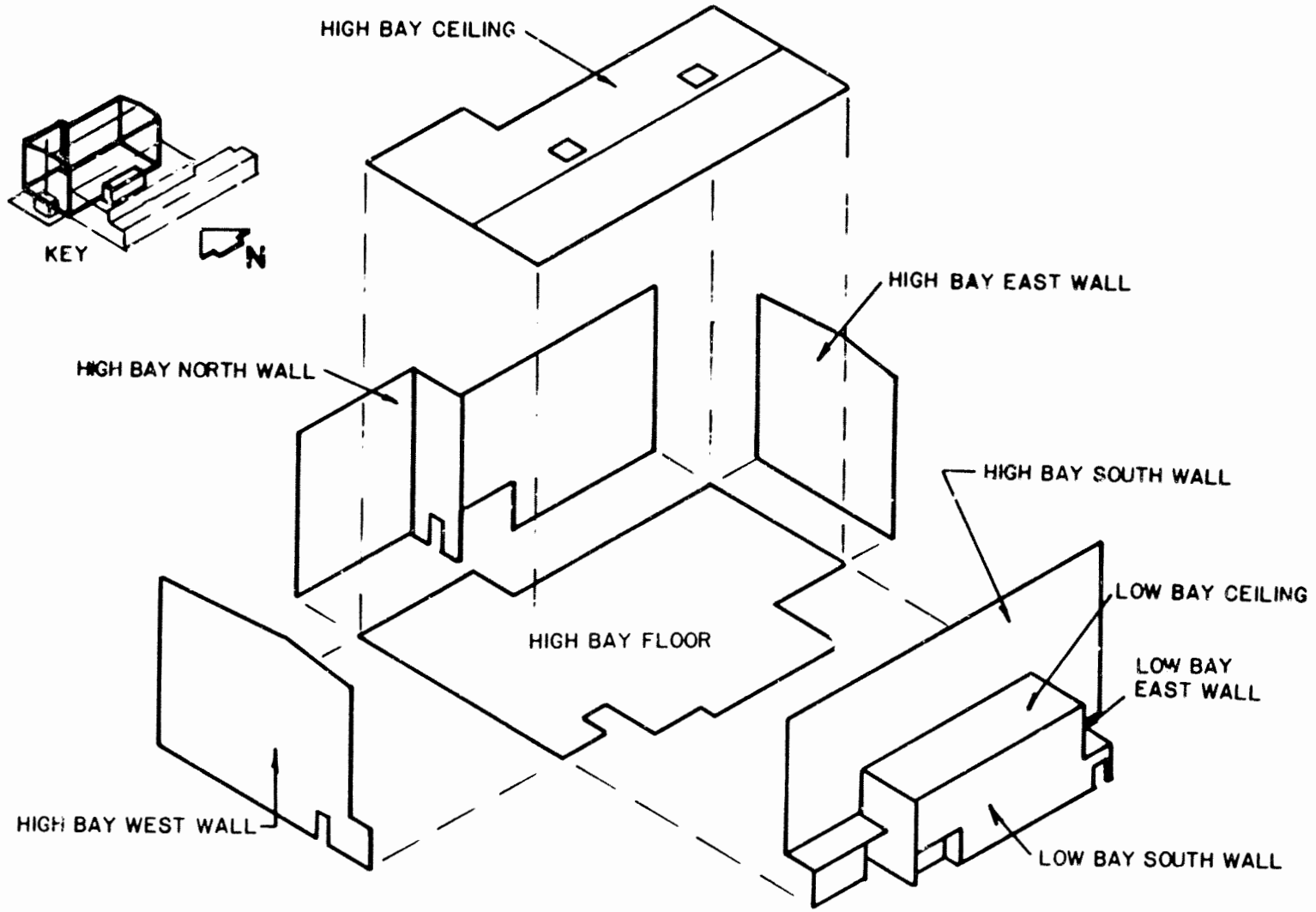
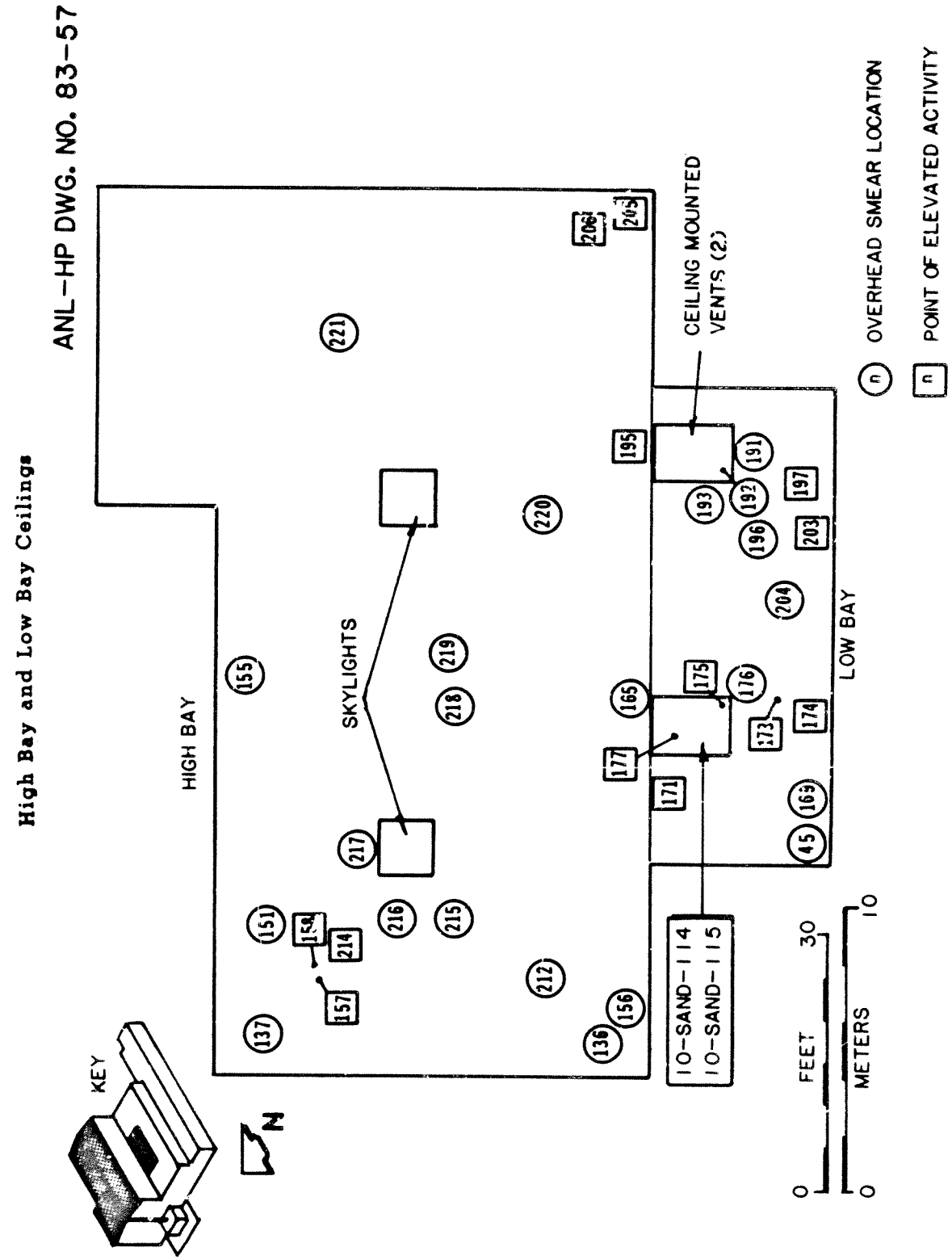


Figure 21

Figure 22


High Bay and Low Bay Ceilings
ANL-HP DWG. NO. 83-57
HIGH BAY
LOW BAY
SKYLIGHTS
CEILING MOUNTED VENTS (2)
10-SAND-114
10-SAND-115
OVERHEAD SMEAR LOCATION
POINT OF ELEVATED ACTIVITY
KEY
N
FEET
METERS
0
30
0
10

High Bay and Low Bay Floors

ANL-HP DWG. NO. 83-58

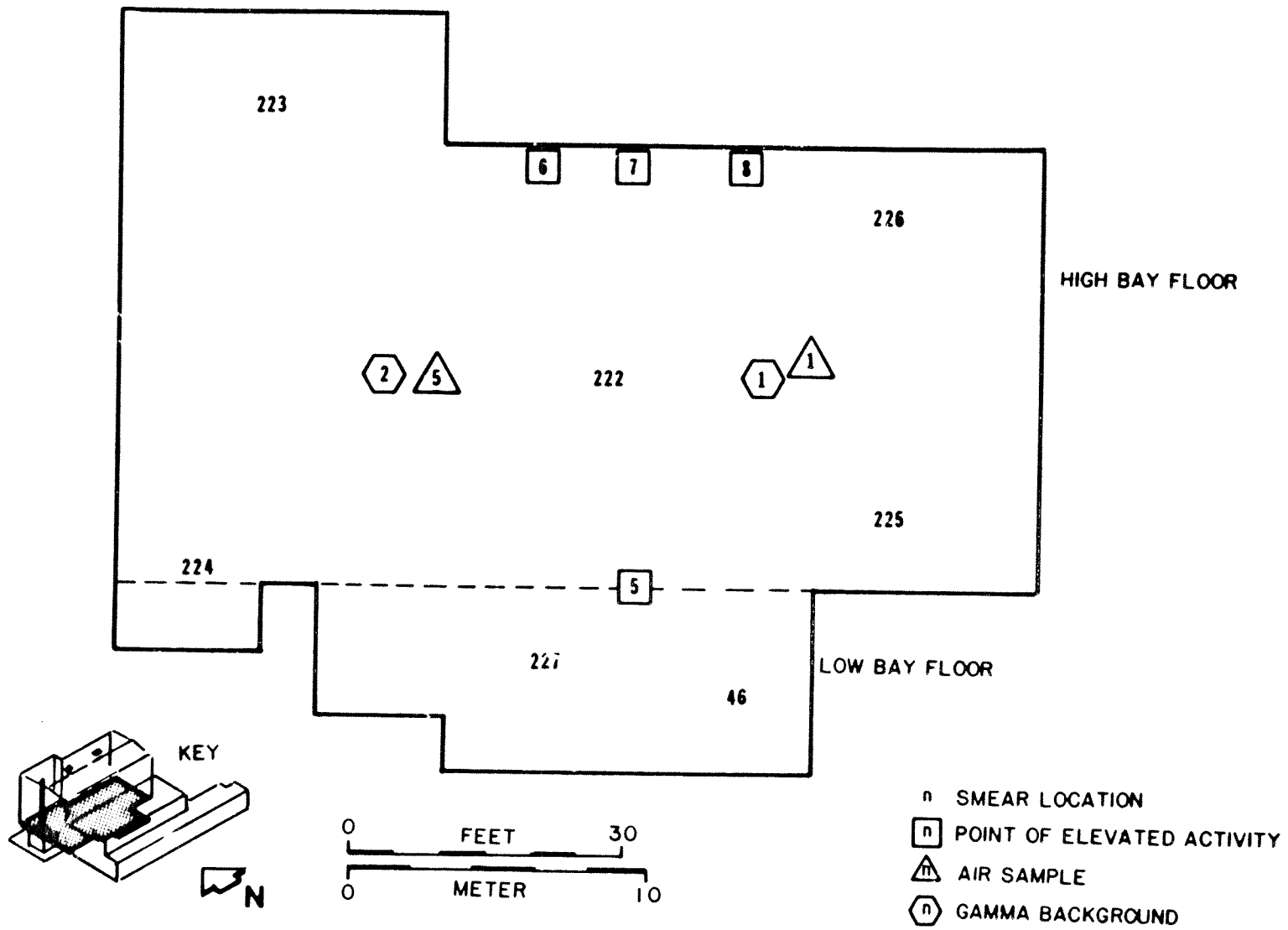
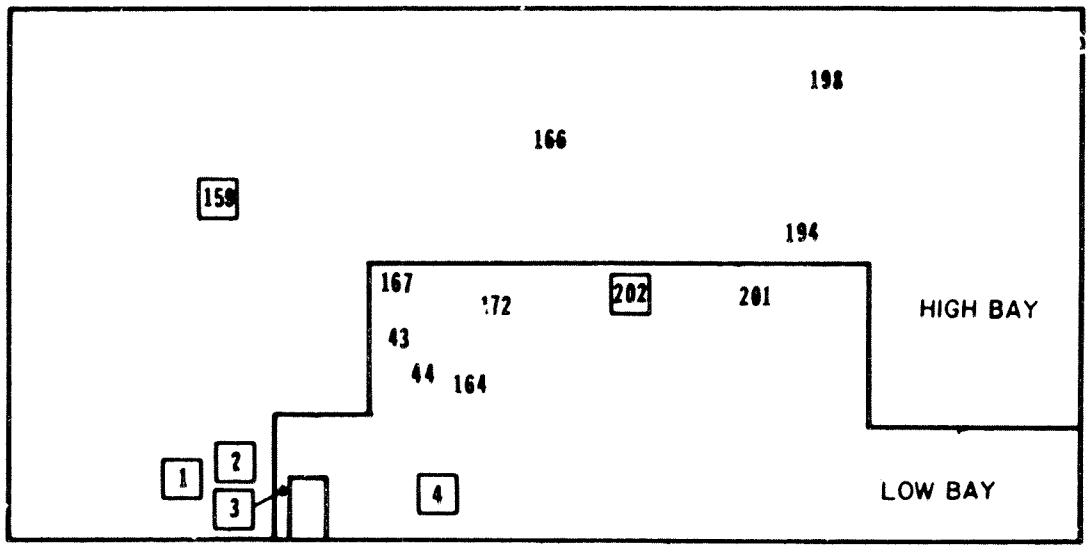
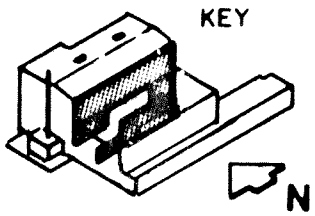


Figure 23

High Bay and Low Bay South Wall

ANL-HP DWG. NO. 83-62





0 FEET 30

0 METERS 10

n SMEAR LOCATION

[n] POINT OF ELEVATED ACTIVITY

Figure 24

High Bay North Wall

ANL-HP DWG. NO. 83-61

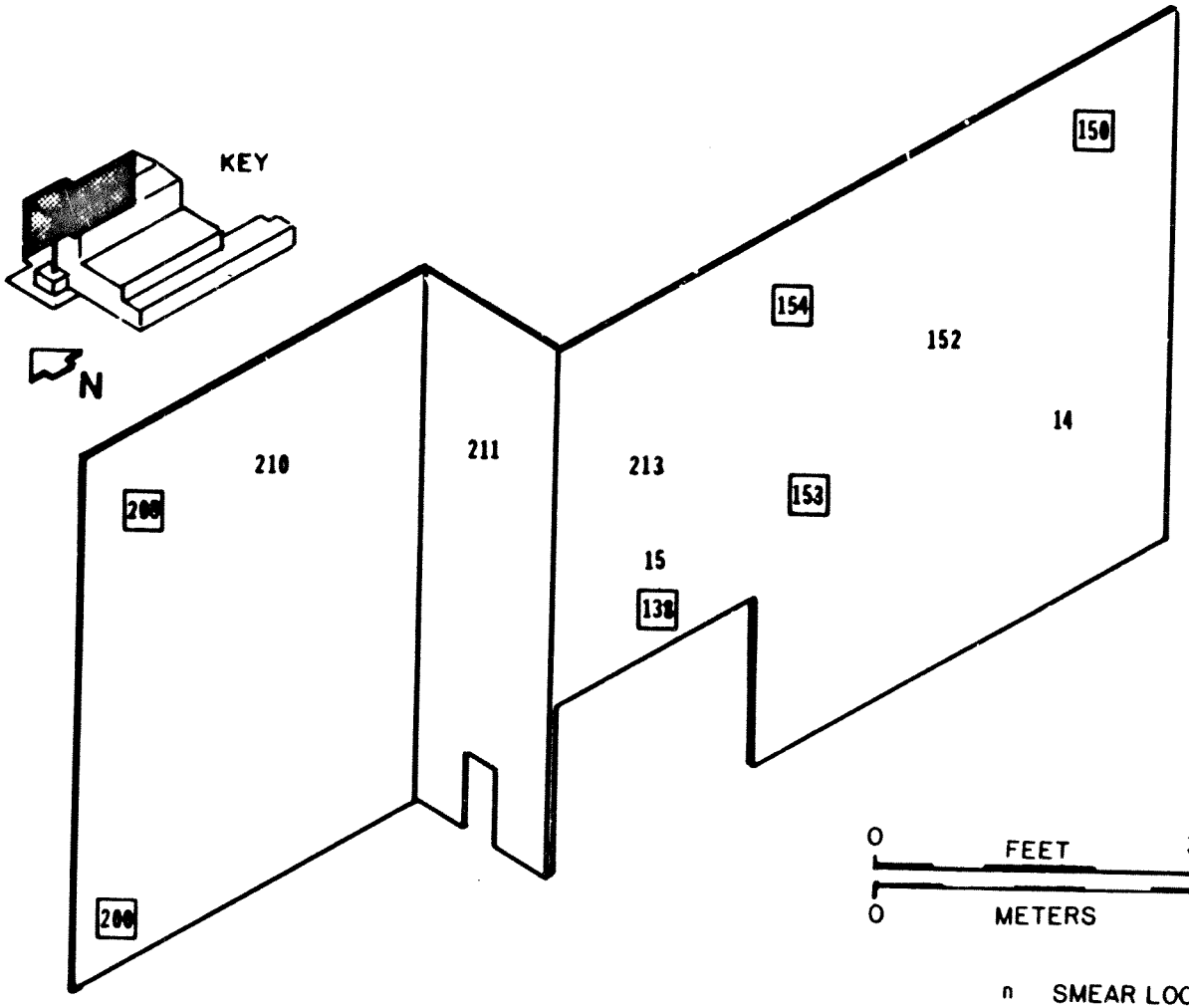
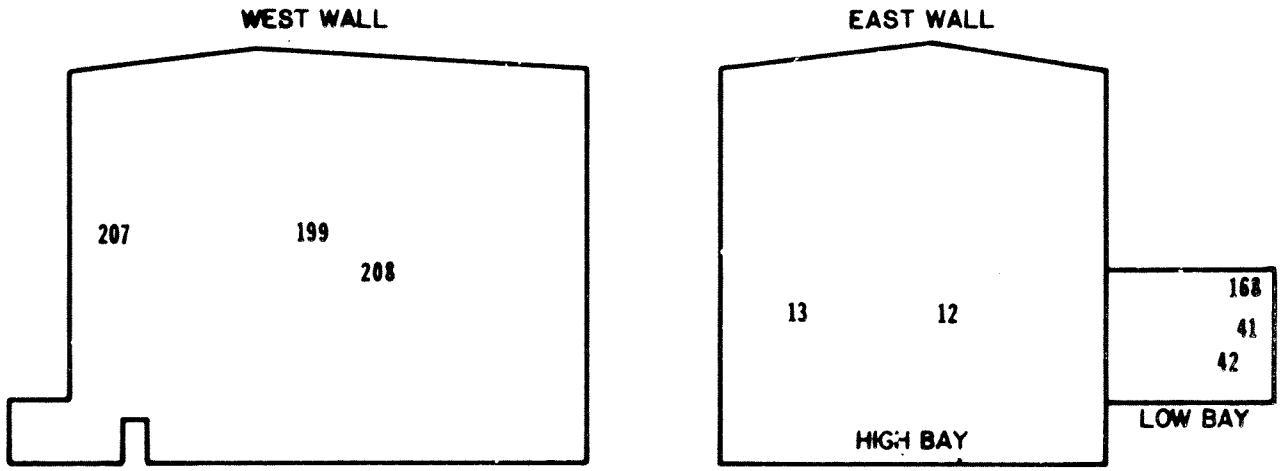
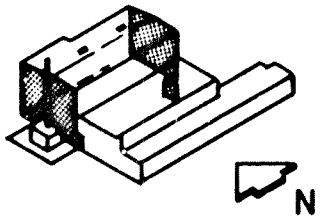


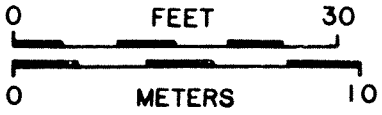
Figure 25

High Bay and Low Bay East and West Walls

ANL–HP DWG. NO. 83–59







n SMEAR LOCATION

Figure 26

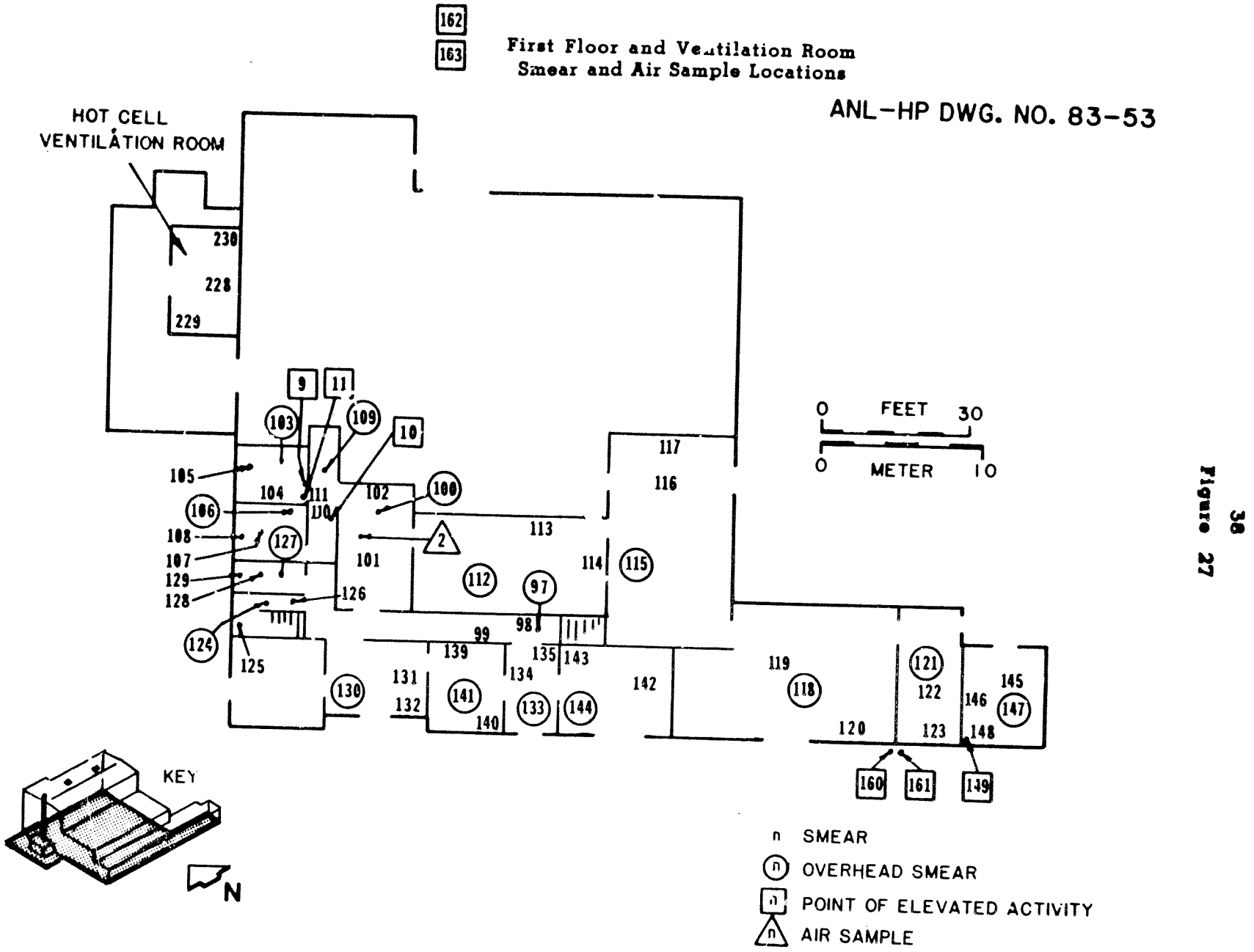

162
163
First Floor and Ventilation Room
Smear and Air Sample Locations
ANL-HP DWG. NO. 83-53
HOT CELL
VENTILATION ROOM
230
228
229
9
11
103
109
10
105
104
111
110
102
100
106
108
127
107
101
129
128
126
124
125
130
131
132
141
140
133
144
139
99
98
135
143
134
142
112
97
114
115
113
117
116
2
0 FEET 30
0 METER 10
119
118
121
122
145
146
147
120
123
148
160
161
149
KEY
N
n SMEAR
(n) OVERHEAD SMEAR
[n] POINT OF ELEVATED ACTIVITY
△n AIR SAMPLE

Figure 27

Roof Smear Locations

ANL-HP DWG. NO. 83-51

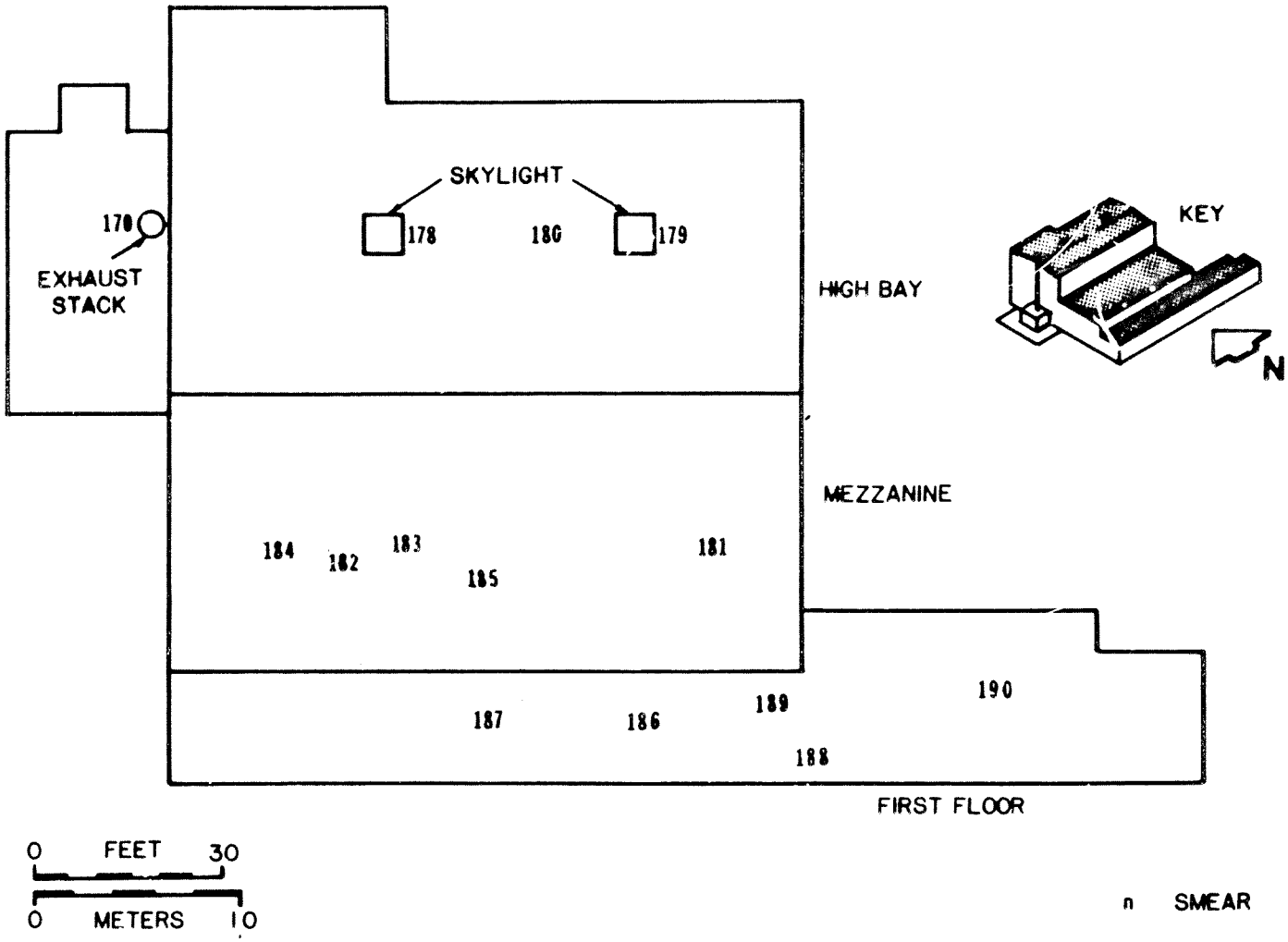


Figure 28

Hot Cell and Mezzanine Smear and Air Sample Locations

ANL-HP DWG. NO. 83-54

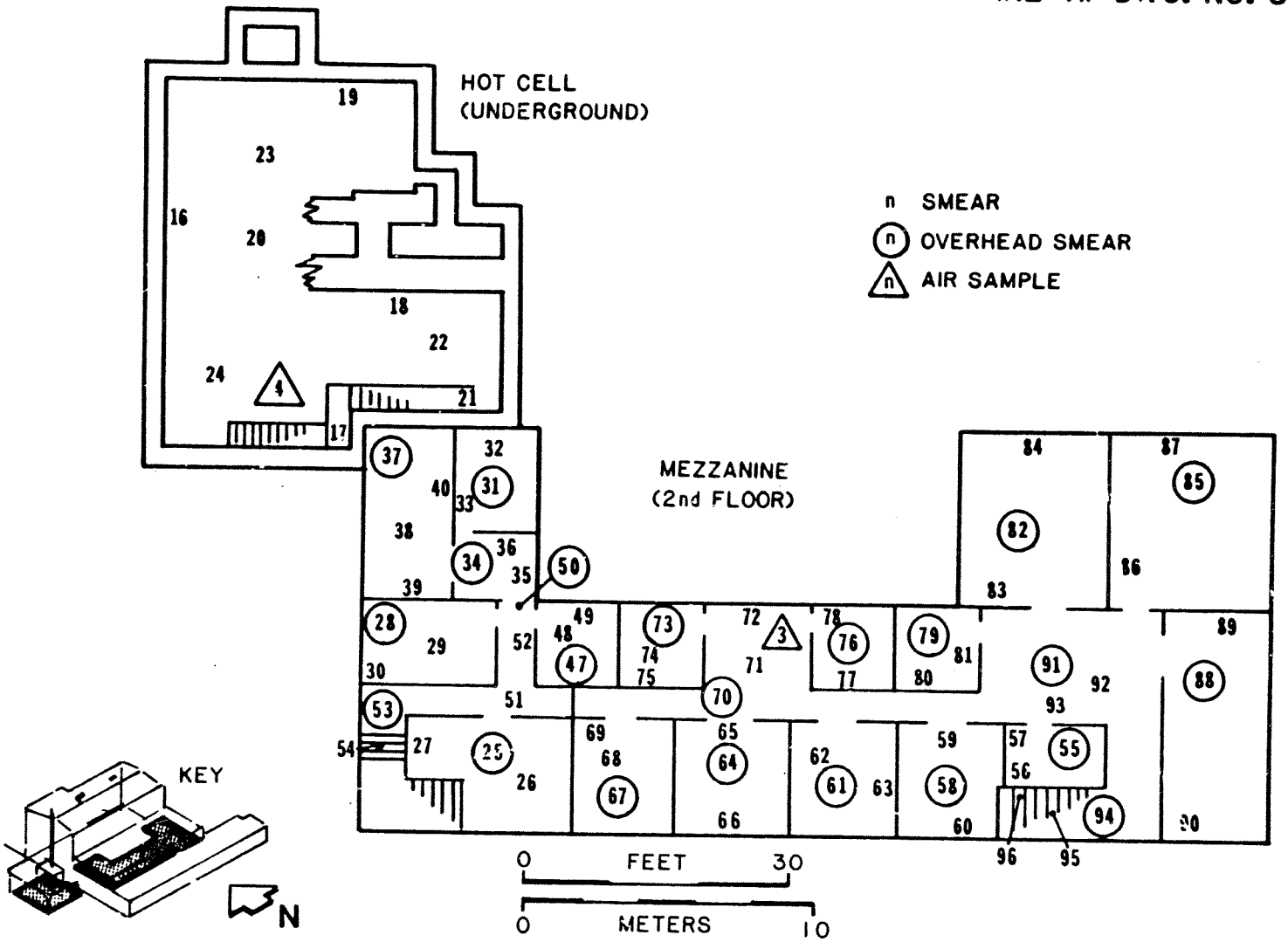


Figure 29

Building 041 - Smear, Air Sample, and Gamma Background Locations

ANL-HP DWG. NO. 83-56

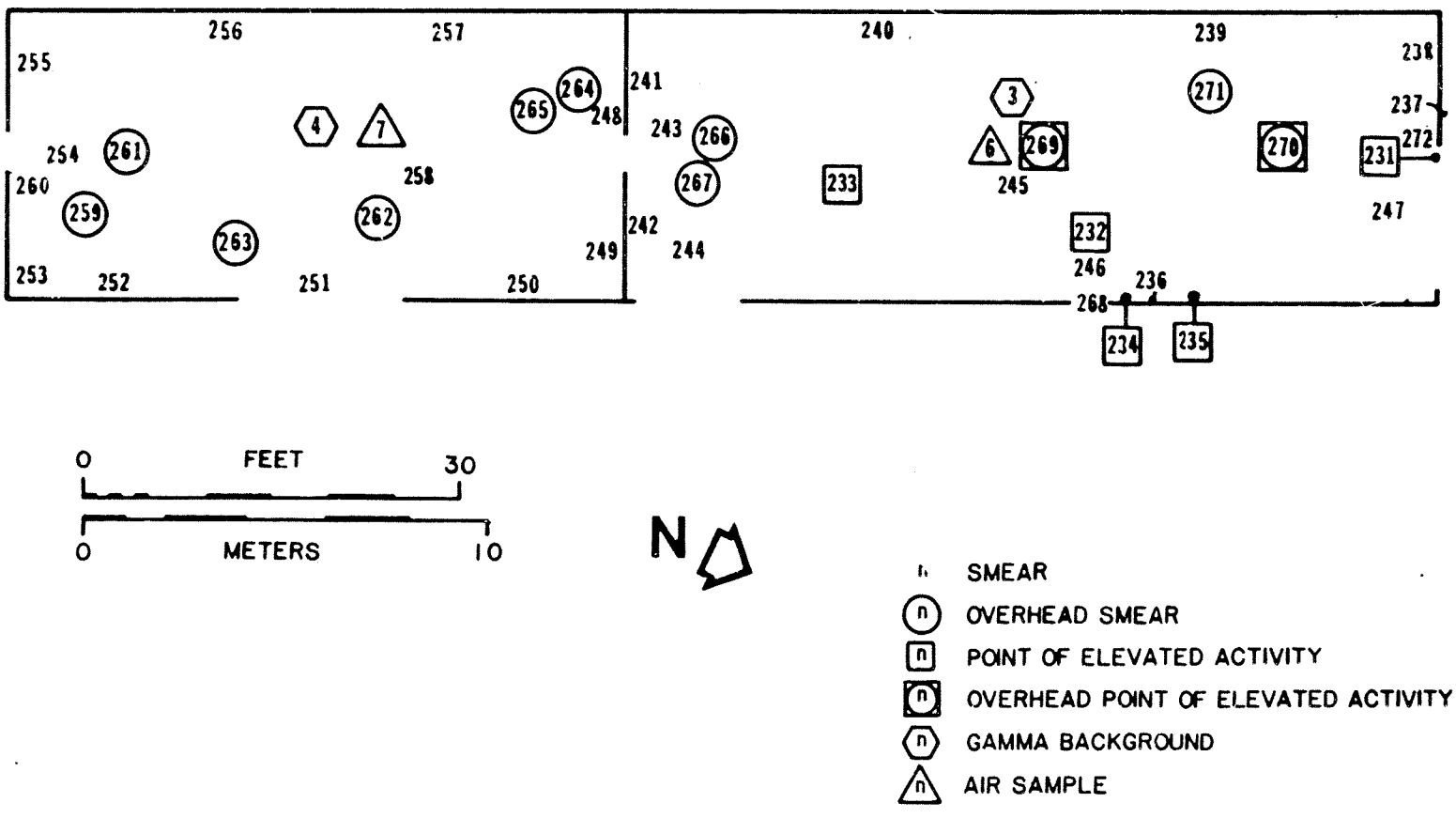
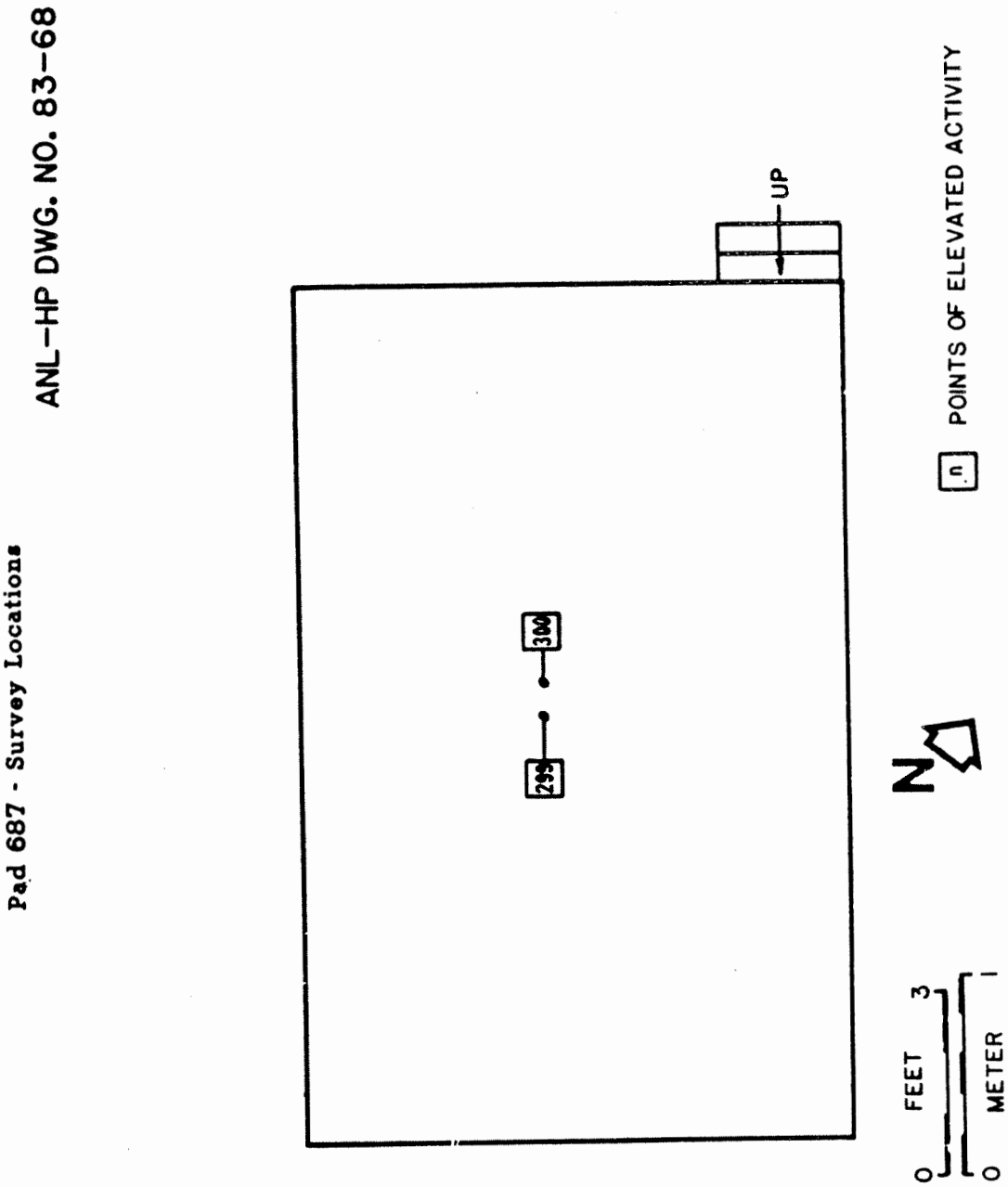


Figure 30

Figure 31


ANL-HP DWG. NO. 83-68
Pad 687 - Survey Locations
UP
300
299
n
POINTS OF ELEVATED ACTIVITY
N
0 FEET 3
0 METER 1

Building 163 - Smear, Air Sample, and Gamma Background Locations

ANL-HP DWG. NO. 83-55

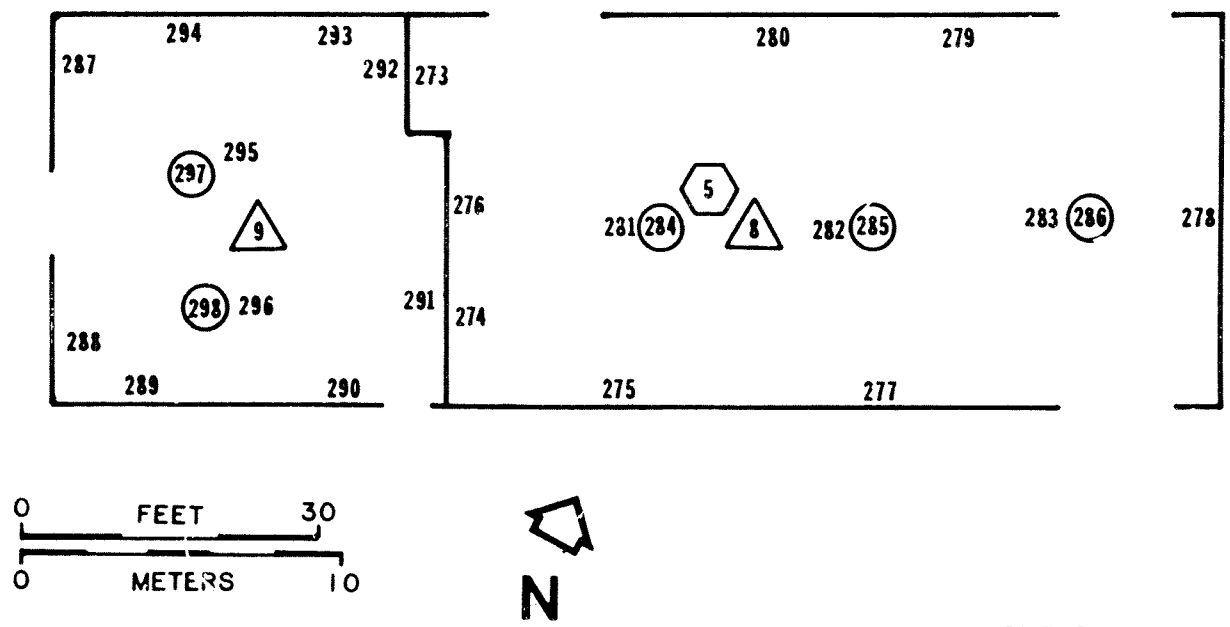


Figure 32

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Percent of Area Accessible for Survey Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
SRE Room 120 South Wall		100								
				2500	BKGD[b]	0.03	NRR[f]	1000	BKGD	Location 1
				2500	BKGD	0.02	NRR	BKGD	BKGD	Location 2
				8000	BKGD	0.03	NRR	BKGD	BKGD	Location 3, Door Frame
				2500	BKGD	BKGD	NRR	BKGD	BKGD	Location 4, Window Ledge
				1000	BKGD	BKGD	NRR	BKGD	BKGD	Location 159, I Beam
				1400	BKGD	NRR	NRR	BKGD	BKGD	Location 202, Fire alarm
				NRR	NRR	NRR	40	NRR	NST[e]	Clean clothes cabinet
				BKGD	NA[c]	NA	BKGD	BKGD	BKGD	All other areas
West Wall		100		BKGD	NA	NA	BKGD	BKGD	BKGD	
North Wall		100		3000	BKGD	NRR	NRR	BKGD	BKGD	Location 138, Beam

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²)		End Window GM (mR/h)	PRM-7 (μR/h)	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter			
				800	BKGD	BKGD	NRR	2500	BKGD	Location 150
				800	BKGD	BKGD	NRR	BKGD	BKGD	Location 153
				BKGD	BKGD	BKGD	NRR	1500	BKGD	Locatio 154
				2500	BKGD	0.035	NRR	3500	BKGD	Locatior 200
				BKGD	BKGD	NRR	NRR	2500	BKGD	Locat:on 209
				BKGD	NA	NA	BKGD	BKGD	BKGD	All other areas
East Wall		100		BKGD	NA	NA	BKGD	BKGD	BKGD	
Floor	100		0.0068 0.0017							Air Sample 1 Air Sample 5
				800	BKGD	0.025	NRR	1000	BKGD	Location 5
				10,000	BKGD	0.035	NRR	4500	BKGD	Location 6, ^{137}Cs-^{137m}Ba
				500	BKGD	0.02	NRR	1500	BKGD	Location 7
				3000	BKGD	0.07	NRR	BKGD	BKGD	Location 8, I Beam
				BKGD	NA	NA	BKGD	BKGD	BKGD	All other areas

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey: Floor	Percent of Area Accessible for Survey: Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²): Beta	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²): Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
Ceiling				5000	BKGD	0.05	NRR	BKGD	BKGD	Location 157, Light fixture
				1000	BKGD	BKGD	NRR	BKGD	BKGD	Location 158, Angle iron
				800	BKGD	BKGD	NRR	BKGD	BKGD	Location 171, I Beam
				1000	BKGD	BKGD	NRR	BKGD	BKGD	Location 173, I Beam
				1000	BKGD	BKGD	NRR	BKGD	BKGD	Location 174, I Beam
				800	BKGD	BKGD	NRR	BKGD	BKGD	Location 175, I Beam
				800	BKGD	0.03	NRR	1500	BKGD	Location 177, Vent unit ^{137}Cs, ^{60}Co, U
				1000	BKGD	0.04	NRR	BKGD	BKGD	Location 195, Vent unit
				7000	BKGD	NRR	NRR	BKGD	BKGD	Location 197
				1400	BKGD	NRR	NRR	BKGD	BKGD	Location 203
				5000	BKGD	0.04	NRR	BKGD	BKGD	Location 205, Beam

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²)		End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
	Floor	Wall		Beta	Alpha					
				3000	BKGD	0.05	NRR	BKGD	BKGD	Location 206, Beam
				400	BKGD	BKGD	NRR	BKGD	BKGD	Location 214, I Beam
				BKGD	NA	NA	NRR	BKGD	BKGD	All other areas
SRE First Floor	80	90	0.0074							Air Sample 2, Room 101
				13,000	BKGD	0.08	NRR	2500	BKGD	Location 9, Door Louver ^{137}Cs-^{137m}Ba
				3000	BKGD	0.05	NRR	1500	BKGD	Location 10, Floor ^{137}Cs-^{137m}Ba
				3000	BKGD	0.04	NRR	2000	BKGD	Location 11, Floor
				26,000	BKGD	0.07	15	4500	BKGD	Location 149, Equated to normal U
				BKGD	NA	NA	BKGD	BKGD	BKGD	Rest of survey

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Result (dis/min-100 cm²)	Comments
SRE Mezzanine			0.0125							Air Sample 3, Room 213
	90	80		BKGD	NA	NA	BKGD	BKGD	BKGD	
SRE Hot Cell	100	100	0.0114	BKGD	NA	NA	BKGD	BKGD	BKGD	Air Sample 4
SRE Hot Cell Exhaust Room	100	100	NS	BKGD	BKGD	NA	NRR	BKGD	BKGD	
SRE Roofs	100		NS	BKGD	NA	NA	BKGD	BKGD	BKGD	
Bldg. 041 North	80	70	0.0005							Air Sample 6
				13,000	BKGD	0.08	NRR	10,000	BKGD	Location 231, Floor
				17,000	BKGD	0.04	NRR	6000	BKGD	Location 232, Floor
				3000	BKGD	0.04	NRR	3000	BKGD	Location 233, Floor

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
				3000	BKGD	BKGD	NRR	BKGD	BKGD	Location 234, Floor
				1000	BKGD	BKGD	NRR	BKGD	BKGD	Location 235, Floor
				BKGD	BKGD	NRR	NRR	2000	BKGD	Location 269, Vent
				BKGD	BKGD	NRR	NRR	2000	BKGD	Location 270, Vent
				BKGD	NA	NA	BKGD	BKGD	BKGD	Rest of survey
Bldg. 041 South	80	70	0.0010	BKGD	NA	NA	BKGD	BKGD		Air Sample 7
Pad 687	70	100	NS	BKGD	NA	NA	BKGD	BKGD	NST	Rest of survey
				61,000	BKGD	NRR	NRR	BKGD	NST	Location 299, Floor
				14,000	BKGD	NRR	NRR	BKGD	NST	Location 300, Floor
Bldg. 163 East	60	20	0.0006	BKGD	NA	NA	BKGD	BKGD		Air Sample 8 Box Shop

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
Bldg. 163 West	50	30	0.0013	BKGD	NA	NA	BKGD	BKGD		Air Sample 9
Bldg. 133			NS	NRR	NA	NA	NRR	5000	NST	Not part of this project
Paved Areas	100		NS							
				30,000	BKGD	0.12	20	7000	BKGD	Location 160, ^{137}Cs, Asphalt
				30,000	BKGD	0.18	20	14,000	BKGD	Location 161, ^{137}Cs, Asphalt
				11,000	BKGD	0.04	NRR	2500	BKGD	Location 162, Asphalt
				11,000	BKGD	0.04	NRR	2300	BKGD	Location 163, Asphalt
				1000	BKGD	BKGD	25	10,000	NST	Northwest corner, Region IX
				BKGD	NA	NA	BKGD	BKGD	NST	All other areas

FOOTNOTES FOR TABLE 1

[a]The beta mode direct readings and alpha mode direct readings were taken with PAC-4G-3 instruments. The beta mode detects both electromagnetic and particulate radiation. If an area indicated a higher count rate than the area background, a beta-mode reading was obtained. The instrument was then switched to the alpha mode, and a reading of the alpha contamination was obtained. In the alpha mode the instrument only responds to particles with high specific ionization, such as alpha particles. The beta-mode readings are compensated for any alpha contribution by subtracting the alpha-mode reading from the beta-mode reading. The area background is subtracted in both the alpha- and beta-mode readings to obtain the net readings.

[b]BKGD = Background. The following are the normal area background readings for each instrument.

	Beta Mode	Alpha Mode
Floor Monitor	1500-2000 cts/min-325 cm^2	0-50 cts/min-325 cm^2
PAC-4G-3	200-400 cts/min-51 cm^2	0-5 cts/min-51 cm^2
PC-5 Counter	19.0 ± 5.9 cts/min*	0.3 ± 0.3 cts/min*
PC-3A Counter	64.0 ± 5.6 cts/min*	0.4 ± 0.3 cts/min*
10-Wire	568.9±16.9 cts/min*	2.9 ± 1.2 cts/min*
PRM-5-3	1000-2000 cts/min	
PRM-7	10-15 µR/h	

GM End Window Detector read 0.01 to 0.03 mR/h at floor level.

[c]NA = Nonapplicable. No contamination was detected above background in the beta mode; therefore, no alpha mode or contact GM End Window survey was necessary.

[d] α = Alpna
$\beta\gamma$ = Beta-gamma
(The beta-gamma readings are compensated for any alpha contamination by subtracting the alpha reading. The background count rate is subtracted in both readings.)

[e]NST = No smear taken.

[f]NRR = No reading recorded.

[g]NS = Not selected, indicating that the room or area was not selected for an air sample. Locations of air samples were chosen on a selected basis throughout the areas surveyed.

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*One standard deviation due to counting statistics.

TABLE 2

LOCATIONS WHERE RESIDUAL CONTAMINATION EXCEEDED ACCEPTABLE LIMITS

Building	Location Number	PAC Readings (dis/min-100 cm^2) Beta-gamma	PAC Readings (dis/min-100 cm^2) Alpha	GM Readings (mR/h)	PRM-7 (μR/h)	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm^2) Beta-gamma	Smear Results (dis/min-100 cm^2) Alpha
SRE - High Bay								
South Wall	1	2500	BKGD[a]	0.03	NRR[b]	1000	BKGD	BKGD
" "	2	2500	BKGD	0.02	NRR	BKGD	BKGD	BKGD
" "	3	8000	BKGD	0.03	NRR	BKGD	BKGD	BKGD
" "	4	2500	BKGD	BKGD	NRR	BKGD	BKGD	BKGD
North Wall	138	3000	BKGD	NRR	NRR	BKGD	BKGD	BKGD
North Wall	200	2500	BKGD	0.035	NRR	3500	BKGD	BKGD
Floor	6	10,000	BKGD	0.035	NRR	4500	BKGD	BKGD
Floor	8	3000	BKGD	0.07	NRR	BKGD	BKGD	BKGD
Ceiling	157	3000	BKGD	0.05	NRR	BKGD	BKGD	BKGD
"	197	7000	BKGD	NRR	NRR	BKGD	BKGD	BKGD
"	205	5000	BKGD	0.04	NRR	BKGD	BKGD	BKGD
"	206	3000	BKGD	0.05	NRR	BKGD	BKGD	BKGD
SRE; First floor	9	13,000	BKGD	0.08	NRR	2500	BKGD	BKGD
" "	10	3000	BKGD	0.05	NRR	1500	BKGD	BKGD
" "	11	3000	BKGD	0.04	NRR	2000	BKGD	BKGD
" "	149	26,000	BKGD	0.07	15	4500	BKGD	BKGD
Bldg. 041	231	13,000	BKGD	0.08	NRR	10,000	BKGD	BKGD
	232	17,000	BKGD	0.04	NRR	6000	BKGD	BKGD
	233	3000	BKGD	0.04	NRR	3000	BKGD	BKGD
	234	3000	BKGD	BKGD	NRR	BKGD	BKGD	BKGD

TABLE 2
(cont'd.)

Building	Location Number	PAC Readings (dis/min-100 cm^2) Beta-gamma	PAC Readings (dis/min-100 cm^2) Alpha	GM Readings (mR/h)	PRM-7 (μR/h)	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm^2) Beta-gamma	Smear Results (dis/min-100 cm^2) Alpha
687	299	61,000	BKGD	NRR	NRR	BKGD	NST[c]	NST
	300	14,000	BKGD	NRR	NRR	BKGD	NST	NST
Paved Areas	160	30,000	BKGD	0.12	20	7000	BKGD	BKGD
	161	30,000	BKGD	0.18	20	14,000	BKGD	BKGD
	162	11,000	BKGD	0.04	NRR	2500	BKGD	BKGD
	163	11,000	BKGD	0.04	NRR	2300	BKGD	BKGD
Northwest Corner Region IX		1000	BKGD	BKGD	25	10,000	NST	NST

[a] BKGD = Background. The background readings for each instrument are given in the footnotes for Table 1.

[b] NRR = No Reading Recorded.

[c] NST = No Smear Taken.

TABLE 3

RADON DETERMINATIONS

Air Sample Number	Location	Figure	dis/min-m^3 Alpha (100-min)	^{222}Rn (pCi/ℓ)	WL[a]	^{220}Rn (pCi/ℓ)	^{219}Rn (pCi/ℓ)
1	High Bay	23	82	0.680	0.0068	0.0102	ND[b]
2	Room 101	27	89	0.742	0.0074	0.0054	ND
3	Room 213	29	153	1.252	0.0125	0.0121	ND
4	Hot Cell	29	167	1.140	0.0114	0.0367	ND
5	High Bay	23	28	0.171	0.0017	0.0095	ND
6	Bldg. 41, North	30	9	0.048	0.0005	0.0040	NS[c]
7	Bldg. 41, South	30	14	0.102	0.0010	0.0021	NS
8	Bldg. 163, East	32	9	0.0587	0.0006	0.0021	NS
9	Bldg. 163, West	32	18	0.132	0.0013	0.0028	NS

[a] A Working Level (WL) is defined as any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. The numerical value of the WL is derived from the alpha energy released by the total decay through RaC' of the short-lived radon daughter products RaA, RaB, and RaC at radioactive equilibrium with 100 pCi of ^{22}Rn per liter of air.

[b] ND = Not Detectable by the positive displacement pump.

[c] NS = Not Selected for actinon determination.

TABLE 4

AMBIENT RADIATION LEVELS

Sample	Figure	Integrated Reading (μR)	Integrated Time (hour)	Rate (μR/h)	Building
1	23	52	4	13	High Bay East
2	23	26	2	13	High Bay West
3	30	58	4.67	12.4	041, North
4	30	27	2	13.5	041, South
5	32	56	5.67	9.9	163, East

TABLE 5

SAMPLE WEIGHTS
(grams)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
		SOIL		
10-S-12A	847	760	673	83
10-S-12B	433	411	377	28
10-S-12C	613	597	528	62
10-S-12D	2551	2490	1914	567
10-S-13A	915	806	713	87
10-S-13B	362	330	309	15
10-S-13C	389	373	359	9
10-S-13D	940	911	805	100
10-S-14A	682	612	520	88
10-S-14B	477	447	399	43
10-S-14C	1017	999	728	265
10-S-14D	698	690	608	77
10-S-29A	1808	1568	704	860
10-S-29B	1889	1691	1061	626
10-S-31	1485	1200	968	226
10-S-32A	1448	1404	868	530
10-S-32B	284	274	198	73
10-S-38	1785	1696	967	723
10-S-39	1264	1198	780	409
10-S-40	1532	1458	877	578
10-S-41	1460	1379	976	396
10-S-42	1949	1889	918	956
10-S-43	1285	1239	811	420
10-S-44	2477	2368	1057	1300
10-S-45	1268	1237	780	456
10-S-46	1499	1417	843	573
10-S-48	2080	1973	1074	892
10-S-49	1788	1723	925	798
10-S-50	2709	2600	1148	1442
10-S-51	1948	1900	1034	856
10-S-52	1782	1642	933	701
10-S-53	1192	1142	682	453
10-S-54	2154	1693	1476	205
10-S-56	1210	1126	910	197
10-S-58	870	853	691	156
10-S-59	1400	1327	825	496

TABLE 5
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
		SOIL (continued)		
10-S-92-24	2297	2294	1991	300
10-S-93-24	2431	2428	2150	274
10-S-94-24	2243	2235	2055	177
10-S-95-24	2360	2352	2095	252
10-S-96-30	1907	1896	1470	417
10-S-97-33	1960	1951	1658	291
10-S-98-32	1784	1776	1312	463
10-S-99-24	1702	1695	1337	356
10-S-100-30	1705	1698	1400	298
10-S-101-32	1864	1858	1570	283
10-S-102-32	1911	1903	1426	476
10-S-103-16.4	574	568	443	120
10-S-104-16.4	645	635	514	121
10-S-105-16.4	578	554	431	122
10-S-106-19.7	475	468	368	100
10-S-107-19.7	480	468	392	76
10-S-108-19.7	512	500	389	103
10-S-109-2.0	35	35	35	0
10-S-109-4.0	32	32	32	0
10-S-109-6.0	32	31	27	4
10-S-110-2.0	26	26	26	0
10-S-110-4.0	23	22	22	0
10-S-110-6.0	16	16	16	0
10-S-111-8.0	23	23	23	0
10-S-112-8.0	15	15	15	0
10-S-130	16	11	11	0
		CONCRETE		
10-C33	550			
10-C35	666			
10-C36	309			
10-C37A	44			
10-C37B	261			

TABLE 5
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
		MISCELLANEOUS		
10-REBAR-34	270			
10-ROCK-47	771			
10-ROCK-57		558	273	280
10-ASPHALT-113	83			
10-SAND-114	168			
10-SAND-115	27			
		SEWER SLUDGE		
10-SS-125	101	81	50	30
10-SS-126	444	413	307	101
10-SS-127	263	230	220	6
10-SS-128	7			
10-SS-129	24			

TABLE 6

GAMMA SPECTRAL AND URANIUM-FLUOROMETRIC ANALYSES OF SAMPLES

Sample No.	Gamma Spectra (pCi/g±σ[a]) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric (µg/g±10%)	Uranium Fluorometric (pCi/g±10%[b])
		BACKGROUND SOIL			
10-S-12A	0.95±0.05	1.50 ±0.19	0.85±0.08	2.1±0.21	1.4 ±0.14
10-S-12B	1.55±0.05	1.24 ±0.13	0.70±0.06	2.0±0.20	1.4 ±0.14
10-S-12C	1.21±0.03	1.17 ±0.09	0.63±0.05	4.4±0.44	3.0 ±0.30
10-S-12D	1.56±0.06	1.66 ±0.20	0.81±0.11	2.4±0.24	1.6 ±0.16
10-S-13A	0.61±0.05	1.56 ±0.17	1.00±0.08	2.6±0.26	1.8 ±0.18
10-S-13B	0.50±0.05	1.49 ±0.21	0.78±0.11	2.4±0.24	1.6 ±0.16
10-S-13C	0.13±0.04	1.38 ±0.17	0.81±0.09	2.9±0.29	2.0 ±0.20
10-S-13D	0.10±0.03	1.53 ±0.14	1.02±0.06	3.3±0.33	2.3 ±0.23
10-S-14A	0.43±0.05	1.18 ±0.21	0.78±0.08	1.9±0.19	1.3 ±0.13
10-S-14B	BDL[c]	1.28 ±0.14	0.64±0.07	1.5±0.15	1.0 ±0.10
10-S-14C	BDL	1.00 ±0.15	0.51±0.07	1.4±0.14	1.0 ±0.10
10-S-14D	BDL	0.73 ±0.21	0.51±0.08	1.4±0.14	1.0 ±0.10
		SOIL			
10-S-29A*	< 0.03	1.64±0.12	2.78±0.19	11.7±1.17	8.0±0.80
10-S-29B	< 0.03	1.05±0.07	1.14±0.08	3.7±0.37	2.5±0.25
10-S-31	680.00±50.00	2.02±0.14	1.38±0.10	3.6±0.36	2.5±0.25
10-S-32A	0.46± 0.04	1.22±0.09	0.96±0.07	3.1±0.31	2.1±0.21
10-S-32B	0.93± 0.07	1.67±0.12	0.75±0.06	2.5±0.25	1.7±0.17
10-S-38	0.49± 0.05	2.73±0.19	2.63±0.18	3.4±0.34	2.3±0.23
10-S-39	0.69± 0.06	1.46±0.10	1.24±0.09	3.1±0.31	2.1±0.21
10-S-40	0.44± 0.04	2.30±0.16	2.08±0.15	2.6±0.26	1.8±0.18
10-S-41	< 0.03	1.29±0.09	1.14±0.08	3.4±0.34	2.3±0.23
10-S-42	0.40± 0.04	1.40±0.10	0.84±0.07	3.1±0.31	2.1±0.21
10-S-43	0.60± 0.06	1.46±0.10	1.11±0.08	3.1±0.31	2.1±0.21
10-S-44	0.30± 0.03	1.36±0.10	1.23±0.09	2.7±0.27	1.8±0.18
10-S-45	0.39± 0.04	1.04±0.07	1.01±0.07	2.9±0.29	2.0±0.20
10-S-46	0.30± 0.03	1.93±0.14	1.57±0.11	2.9±0.29	2.0±0.20
10-S-48	110.00±10.00	1.57±0.11	1.26±0.09	2.7±0.27	1.8±0.18
10-S-49	0.19± 0.04	1.33±0.09	1.01±0.07	3.7±0.37	2.5±0.25
10-S-50	0.34± 0.03	1.37±0.10	1.32±0.09	2.6±0.26	1.8±0.18
10-S-51	0.67± 0.06	1.51±0.11	0.94±0.07	3.1±0.31	2.1±0.21
10-S-52	0.03± 0.02	1.34±0.09	2.06±0.14	4.6±0.46	3.1±0.31
10-S-53	< 0.03	1.12±0.08	1.09±0.08	2.4±0.24	1.6±0.16
10-S-54	0.07± 0.03	1.36±0.10	1.16±0.08	3.3±0.33	2.3±0.23
10-S-56	0.18± 0.04	1.70±0.12	1.38±0.10	3.3±0.33	2.3±0.23

TABLE 6
(cont'd.)

Sample No.	Gamma Spectra (pCi/g±σ[a]) ^{137}Cs	^{232}Th Decay Chain	^{226}Th Decay Chain	Uranium Fluorometric (μg/g±10%)	(pCi/g±10%)[b]
10-S-58	< 0.03	1.01±0.07	1.09±0.08	2.4±0.24	1.6 ±0.16
10-S-59	< 0.03	1.48±0.10	1.55±0.11	3.3±0.33	2.3 ±0.23
10-S92-24	< 0.03	1.66±0.17	1.23±0.12	2.9±0.29	2.0 ±0.20
10-S93-24	< 0.03	2.12±0.21	1.79±0.18	4.0±0.40	2.7 ±0.27
10-S94-24	< 0.03	1.71±0.17	1.13±0.11	2.6±0.26	1.8 ±0.18
10-S95-24	< 0.03	1.46±0.15	1.37±0.14	2.9±0.29	2.0 ±0.20
10-S96-30	< 0.03	1.73±0.17	1.11±0.11	2.8±0.28	1.9 ±0.19
10-S97-33	< 0.03	2.63±0.26	1.74±0.17	4.3±0.43	2.9 ±0.29
10-S98-32	BDL	1.74±0.17	1.24±0.12	2.9±0.29	2.0 ±0.20
10-S99-24	BDL	1.91±0.19	1.50±0.15	3.3±0.33	2.3 ±0.23
10-S100-30	BDL	1.83±0.18	1.20±0.12	2.9±0.29	2.0 ±0.20
10-S101-32	BDL	1.88±0.19	1.40±0.14	2.9±0.29	2.0 ±0.20
10-S102-32	BDL	1.80±0.18	1.58±0.16	2.9±0.29	2.0 ±0.20
10-S103-16.4	< 0.03	1.82±0.18	1.50±0.15	2.6±0.26	1.8 ±0.18
10-S104-16.4	BDL	1.54±0.15	1.24±0.12	2.8±0.28	1.9 ±0.19
10-S105-16.4	< 0.03	1.56±0.16	1.33±0.13	2.5±0.25	1.7 ±0.17
10-S106-19.7	< 0.03	1.52±0.15	1.77±0.18	2.6±0.26	1.8 ±0.18
10-S107-19.7	BDL	1.50±0.15	1.26±0.13	2.6±0.26	1.8 ±0.18
10-S108-19.7	BDL	1.70±0.17	1.35±0.14	2.7±0.27	1.8 ±0.18
10-S109-2.0	< 0.03	1.96±0.20	1.31±0.13	2.9±0.29	2.0 ±0.20
10-S109-4.0	3.09±0.31	2.09±0.21	1.21±0.12	2.4±0.24	1.6 ±0.16
10-S109-6.0	< 0.03	1.37±0.14	1.23±0.12	2.7±0.27	1.8 ±0.18
10-S110-2.0	< 0.03	1.53±0.15	0.84±0.08	2.6±0.26	1.8 ±0.18
10-S110-4.0	0.51±0.05	1.33±0.13	1.29±0.13	2.1±0.21	1.4 ±0.14
10-S110-6.0	< 0.03	1.56±0.16	1.15±0.12	1.6±0.16	1.1 ±0.11
10-S111-8.0	BDL	1.07±0.11	0.85±0.08	2.0±0.20	1.4 ±0.14
10-S112-8.0	0.32±0.05	1.27±0.13	1.23±0.12	2.8±0.28	1.9 ±0.19
10-S-130	0.18±0.05	1.10±0.11	0.78±0.08	2.9±0.29	2.0 ±0.20

TABLE 6
(cont'd.)

Sample No.	Gamma Spectra (pCi/g±σ[a]) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric (µg/g±10%)	Uranium Fluorometric (pCi/g±10%)[b]
		CONCRETE			
10-C33	0.14± 0.04	1.23±0.09	0.61±0.05	2.8 ±0.28	1.9 ±0.19
10-C35	0.19± 0.04	<0.06	0.46±0.04	1.5 ±0.15	1.0 ±0.10
10-C36	0.05± 0.03	0.49±0.10	0.37±0.03	1.2 ±0.12	0.82±0.08
10-C37-A	2560.00±180.00	<0.06	<0.02	1.7 ±0.17	1.2 ±0.12
10-C37-B	490.00± 35.00	0.10±0.04	0.68±0.06	1.1 ±0.11	0.75±0.08
		WATER & SEWER SLUDGE			
10-SS-125	4.31± 0.43	1.00±0.10	1.14±0.11	2.2 ±0.22	1.5 ±0.15
10-SS-126	6.19± 0.62	0.94±0.09	0.90±0.09	2.0 ±0.20	1.4 ±0.14
10-SS-127	1.22± 0.12	1.05±0.10	0.80±0.08	1.6 ±0.16	1.1 ±0.11
		Suspended Solids			
10-W30	<0.03	<0.06	<0.02	-	-
10-W55	<0.03	<0.06	<0.06	-	-
10-SS128	2.00± 1.00	0.30±0.20	<0.02	1.4 ±0.1	0.9 ±0.1
10-SS129	3.50± 2.00	3.30±2.00	1.00±0.40	4.9 ±0.5	3.4 ±0.3
		Dissolved Solids		µg/mℓ	pCi/mℓ
10-W-30	<0.03	<0.06	<0.02	-	-
10-W-55	<0.03	<0.06	<0.02	-	-
10-SS128	0.06± 0.03	<0.06	<0.02	0.003	0.002
10-SS129	0.12± 0.03	<0.06	0.05±0.02	0.006	0.004
		OTHERS			
				µg/g	pCi/g
10-REBAR-34	< 0.03	<0.06	<0.02	-	-
10-ROCK-47	0.22± 0.04	1.38±0.10	1.36±0.10	2.40±0.24	1.60±0.16
10-ROCK-57	< 0.03	1.14±0.08	1.41±0.10	4.00±0.40	2.70±0.27
10-ASPHALT-113	786.00±79.00	BDL[c]	BDL	1.00±0.10	0.68±0.07
10-SAND-114	23.40± 2.30	0.25±0.08	0.11±0.03	0.41±0.04	0.28±0.03
10-SAND-115	772.00±77.00	0.46±0.07	0.88±0.09	1.70±0.17	1.20±0.12

TABLE 6
(cont'd.)

[a]One standard deviation due to counting statistics.

[b]ANL conversion from Appendix 5.

[c]BDL = Below Detectable Levels.

*Mass spectrometric analysis yielded normal uranium isotopic ratios (^{234}U - 0.004% ± 0.002%, ^{235}U - 0.724% ± 0.007%, ^{238}U - 99.272% ± 0.02%).

TABLE 7

INDUCED RADIOACTIVITY CONCENTRATIONS OF SAMPLES

Sample No.	Gamma Spectra (pCi/g±σ[a]) ^{137}Cs	^{60}Co	^{152}Eu	Chemical Separation & Gross Beta ^{90}Sr (pCi/g±σ)
		SOIL		
10-S-12C	1.21± 0.03	0.03±0.02	BDL[b]	
10-S-13D	BKGD	0.05±0.03	BDL	
10-S29A	BKGD	BDL	BDL	1.39± 0.32
10-S31	680.00± 50.00	4.76±0.33	BDL	4490.00±1050.00
10-S32B	BKGD	2.60±0.18	25.70±2.60	
10-S44	BKGD	BDL	0.43±0.08	
10-S48	110.00± 10.00	BDL	BDL	8.40± 1.28
10-S109-4.0	3.09± 0.31	BDL	BDL	
10-S130	BKGD	BDL	BDL	4.82± 0.50
		CONCRETE		
10-C33-1	BKGD	2.80±0.20	25.80±2.60	1.23± 0.16
10-C37-A	2560.00±180.00	BDL	BDL	
10-C37-B	490.00± 35.00	BDL	BDL	5.10± 0.90
		SEWER SLUDGE		
10-SS-125	4.31± 0.43	0.15±0.03	BDL	0.80± 0.08
10-SS-126	6.19± 0.62	0.03±0.02	BDL	0.67± 0.07
10-SS-127	1.22± 0.12	BDL	BDL	0.25± 0.03
		Suspended Solids		
10-SS-128	2.00± 1.00	BDL	BDL	1.42± 0.18[d]
10-SS-129	3.50± 2.00	BDL	BDL	1.11± 0.12[d]
		Dissolved Solids		pCi/mℓ
10-SS128	BKGD[c]	BDL	BDL	4.70± 0.50[d]
10-SS129	BKGD	BDL	BDL	0.03± 0.01[d]
		MISCELLANEOUS		
10-REBAR-34	BKGD	40.50±8.10	BDL	
10-ASPHALT-113	786.00± 79.00	BDL	BDL	
10-SAND-114	23.40± 2.30	0.44±0.07	BDL	
10-SAND-115	772.00± 77.00	50.00±5.00	BDL	

TABLE 7
(cont'd.)

[a]One standard deviation due to counting statistics.

[b]BDL = Below Detectable Levels.

[c]BKGD = Background level. (The actual concentrations of ^{137}Cs are given in Table 6.)

[d]Reagent blank including strontium carrier 0.07±0.12.
Data not corrected for this blank.
Aqueous portion weights are based on original solid weights.

TABLE 8

CHEMICAL SEPARATION AND ALPHA SPECTROMETRIC ANALYSIS OF SAMPLES
(pCi/g±σ)

Sample No	$^{239,240}Pu$	^{241}Am	^{228}Th	^{230}Th	^{232}Th
		SOIL			
10-S29A	0.004 ±0.002	0.006±0.003	1.00±0.10	2.57±0.15	0.96±0.10
10-S31	0.039 ±0.005	0.006±0.003	1.40±0.12	1.43±0.13	1.30±0.12
10-S48	0.001 ±0.001	0.001±0.001	0.85±0.09	1.22±0.11	0.87±0.09
10-S58	0.004 ±0.002	< 0.05 ±0.02*	0.56±0.05	0.60±0.05	0.51±0.05
10-S59	0.001 ±0.001	< 0.05 ±0.02*	1.03±0.05	0.95±0.05	0.97±0.05
		CONCRETE			
10-C33-1	0.002 ±0.002	0.001±0.001	0.72±0.10	0.60±0.10	0.70±0.10
10-C37-B	0.006 ±0.002	< 0.05 ±0.02*	0.10±0.03	0.63±0.05	0.09±0.03
		ROCK			
10-R57	0.004 ±0.002	< 0.05 ±0.02*	1.06±0.10	1.05±0.10	1.01±0.10

*Higher values result from correction for unusually high ^{241}Am values in reagent-processing blank samples.

APPENDIX 1

INSTRUMENTATION

I. PORTABLE RADIATION SURVEY METERS

A. Gas-Flow Proportional Survey Meters

The Eberline PAC-4G-3 was the primary instrument used for surveying. This instrument is a gas-flow proportional counter which utilizes a propane gas-proportional detector, 51 cm^2 (AC-21), 100 cm^2 or 325 cm^2 (AC-22) in area, with a thin double-aluminized Mylar window (~ 0.85 mg/cm^2).

Since this instrument has multiple high-voltage settings, it can be used to distinguish between alpha and beta-gamma contamination. This instrument was initially used in the beta mode. In that mode, the detector responds to alpha and beta paticles and x- and gamma-rays. When areas indicated a higher count rate than the average instrument background, the beta-mode reading was recorded, and the instrument was then switched to the alpha mode to determine any alpha contribution. In the alpha mode, the instrument responds only to particles with high specific ionization.

The alpha voltage is set to 1600 V, and the input discriminator is set to 1.5 mV. The instrument is then calibrated in the alpha mode with four flat-plate, infinitely-thin, NBS-traceable ^{239}Pu standards, and in the beta mode with a flat-plate, infinitely-thin NBS-traceable ^{90}Sr-^{90}Y standard. The PAC-4G-3 instruments are calibrated to an apparent 50% detection efficiency.

B. Beta-Gamma End Window Survey Meter

When an area of contamination was found with a PAC instrument, a reading was taken with an Eberline Beta-gamma Geiger-Mueller Counter, Model E-530, with a HP-190 probe. This probe has a thin mica end window and is, therefore, sensitive to alpha and beta particles and x- and gamma-rays. A thin piece of aluminum is added to the mica, making the window density approximately 7 mg/cm^2. At this density, the instrument is not sensitive to the majority of alpha emissions. A maximum reading is obtained with the probe placed in contact with the area of contamination. In this position, the response (in mR/h) to gamma radiation is generally conservative relative to a determination of mrad/h at 1 cm. This instrument is calibrated with an NBS traceable ^{226}Ra source.

C. Low-Energy Gamma Scintillation Survey Meter

An Eberline Model PRM-5-3 with a PG-2 gamma scintillation detector was used to determine low-energy x and gamma radiation. The PG-2 detector consists of a thin NaI(Tℓ) scintillation crystal 5 cm in diameter by 2 mm thick. This instrument is calibrated on three separate discriminators for three energy regions using ^{239}Pu (17 keV), ^{241}Am (59.5 keV) and ^{235}U (185.7 keV) sources. This instrument can be operated in either a differential (to discriminate between different energy regions) or integral mode.

APPENDIX 1
(cont'd.)

D. High Energy Micro "R" Scintillation Survey Meter

An Eberline Micro "R" meter model PRM-7 was used to detect high-energy gamma radiation. This instrument contains an internally mounted NaI(Tℓ) scintillation crystal 2.5 cm in diameter by 2.5 cm thick and can be used to measure fields of low-level radiation between 10 μR/h and 5000 μR/h. This instrument is used to survey ambient background radiation. It is held 1 m from the surface during the survey. This instrument is also calibrated with an NBS standard ^{226}Ra source.

E. Integrating Radiation Meter

In addition to the PRM-7, a pressurized ion chamber (Reuter Stokes Model RSS-111) was used at selected locations to determine the ambient radiation field. The RSS-111 has three output modes: (1) instantaneous exposure rate, (2) strip chart differential readout, and (3) integrated exposure. The chamber is mounted on a tripod, 3 ft (~ 1 m) above the surface and has a uniform energy response from about 0.2 MeV to about 4 MeV. A 2-hour period of operation is usually sufficient to obtain significant data.

II. SMEAR-COUNTING INSTRUMENTATION

An ANL-designed gas-flow proportional detector connected to an Eberline Mini Scaler Model MS-2 was used to count multiple smears simultaneously. This detector has a double-aluminized Mylar window (400 cm^2) and uses P-10 (90% argon and 10% methane) as the counting gas. The metal sample holder for this detector has been machined to hold ten smear papers. This particular system consists of two Mini Scalers and two detectors. One is used to count in the alpha mode; the other is used in the beta mode. Up to ten samples can be counted simultaneously.

Any smear taken from a contaminated area was counted individually in a Nuclear Measurements Corporation gas-flow proportional counter (PC-5 or PC-3A). These instruments have been modified to contain a double-aluminized Mylar spun top (window). This top is placed over non-conducting media (e.g. paper smears) to negate the dielectric effect on the counter. These counters also use P-10 counting gas. Smears are counted in both the alpha and beta modes.

The PC counters are calibrated by adjusting the input discriminator with the high voltage set at 700 V until it begins to count an alpha source. The plateaus are run to establish the operating voltages for alpha and beta-gamma. The MS-2 input discriminator is set to 2 mV and again plateaus are run to establish the operating voltages.

III. AIR-SAMPLING DEVICE

Air samples were collected using a commercially available (ANL-modified filter queen) vacuum cleaner identified as the "Princess Model." Air was drawn through a filter medium at a flow rate of 40 m^3/h. The filters consist of

APPENDIX 1
(cont'd.)

200-cm^2 sheets of Hollingsworth-Vose (HV-70 or LB5211-9 mil) filter paper. The collection efficiency at these flow rates for 0.3-μm particles is about 99.9%.

A separate air sample can be taken with a positive displacement pump drawing about 20 liters/min through a Millipore membrane (0.5 to 0.8 μm) filter paper for about one hour. An alpha spectrum can be measured from a section of this filter paper. The ratio of actinon (^{219}Rn - 6.62 MeV α AcC) to radon (^{222}Rn - 7.69 MeV RaC') can be determined from this spectrum.

IV. GAMMA SPECTRAL INSTRUMENTATION

A Nuclear Data Multichannel Analyzer Model ND-100, utilizing a 7.6-cm-diameter by 7.6-cm-thick NaI(Tℓ) scintillation crystal is commonly used for determining a gamma spectrum. This instrument is calibrated with NBS-traceable gamma sources. Samples from contaminated areas were analyzed using this system, and the contamination radionuclides were identified.

Hyperpure Germanium detectors (ORTEC - 17% efficiency right-circular cylinders) can be used when more sophisticated gamma-ray analyses are required. These detectors are coupled to Nuclear Data Multichannel Analyzers (Models ND-60, ND-66 or ND-100).

APPENDIX 2

CONVERSION FACTORS

I. INSTRUMENTATION

The factors used to convert the instrument readings to units of disintegrations per minute per 100 cm^2 (dis/min-100 cm^2) and the derivation of those factors are listed below.

A. Conversion Factors

	PAC-4G-3 (51 cm^2)		PAC-4G-3 (100 cm^2)		Floor Monitor (FM-4G)	
	Alpha	Beta	Alpha	Beta	Alpha	Beta
To 100 cm^2	1.96	1.96	1.00	1.00	0.31	0.31
dis/min per cts/min for ^{90}Sr-^{90}Y	-	2	-	2	-	2
dis/min per cts/min for ^{239}Pu	2	-	2	-	2	-
dis/min per cts/min for normal U	3.4	2.7	2.7	2.4	2.9	2.5
dis/min per cts/min for ^{226}Ra plus daughters	1.7	1.7	1.7	1.8	1.7	1.8

B. Derivation of Conversion Factors

. Floor Monitor

Window Area: ~ 325 cm^2
Conversion to 100 cm^2 = 0.31 times Floor Monitor readings

. PAC-4G-3

Window Area: ~ 51 cm^2
Conversion to 100 cm^2 = 1.96 times PAC reading

Window Area: ~ 100 cm^2
Conversion factor = 1.00 times PAC reading

APPENDIX 2
(cont'd)

2π Internal Gas-Flow Counter, PC counter

Geometry: Solid Steel Spun Top - 0.50

Geometry: Mylar Spun Top - 0.43
Mylar spun top counting {double-aluminized Mylar window (~ 0.85 mg/cm^2)} utilizes the well of the PC counter and is a method developed and used by the Argonne National Laboratory Health Physics Section for negating the dielectric effect in counting samples on nonconducting media.

The PAC-4G-3 and PC counter were calibrated as described in Appendix 1. Using a flat-plate, infinitely thin ^{226}Ra plus short-lived daughters standard as a source of alpha emissions, the plate was counted in a 2π Internal Gas-Flow Counter (PC counter) with the source leveled to an apparent 2π geometry. The alpha counts per minute (cts/min) reading was found to be 1.86×10^4 cts/min, or $1.86 \times 10^4 \div 0.51^* = 3.65 \times 10^4$ disintegrations per minute (dis/min) alpha. Since the source was infinitely-thin, the alpha component was used as the total alpha dis/min of the source.

The same ^{226}Ra plus daughters source, when counted with the PAC instrument in the alpha mode, was found to be 2.18×10^4 cts/min at contact. The conversion factor for cts/min to dis/min for the PAC instrument is $3.65 \times 10^4 \div 2.18 \times 10^4 = 1.7$ dis/min alpha per cts/min alpha.

The same source was covered with two layers of conducting paper, each 6.31 mg/cm^2, to absorb the alpha emissions. With the PC counter in the beta mode and the paper in good contact with the chamber, the count was found to be 1.17×10^4 or $1.17 \times 10^4 \div 0.50 = 2.35 \times 10^4$ dis/min beta. With the PAC-4G-3 in the beta mode and in contact with the covered source in the center of the probe, the count was found to be 1.36×10^4 cts/min. This indicates a conversion factor of $2.35 \times 10^4 \div 1.36 \times 10^4 = 1.7$ dis/min per cts/min beta-gamma. All three detectors (51 cm^2, 100 cm^2 and 325 cm^2) gave readings similar to those reported above for the alpha and beta-gamma modes.

Utilizing a 1.25 in x 1.25 in x .005 in (3.2 cm x 3.2 cm x 0.013 cm) normal uranium foil as a source of uranium alpha emissions, the foil was counted in a PC counter with the source leveled to an apparent 2π geometry. The same normal uranium source, covered with two layers of conducting paper in good contact with the chamber, each 6.31 mg/cm^2 to negate the alpha emissions, was counted for composite beta and gamma emissions in the PC counter. The source was leveled to an apparent 2π geometry; however, no provision was made for backscatter.

*The value of 0.51 includes the following factors: geometry (g) = 0.50; backscatter factor (bf) = 1.02; sample absorption factor (sa) = 1.0; window air factor (waf) = 1.0. The product of g x bf sa x waf is 0.51.

APPENDIX 2
(cont'd.)

The normal uranium source was also counted with the PAC instrument using all three detector areas in the alpha mode and covered with two layers of conducting paper in the beta mode. The conversion factors were calculated as for ^{226}Ra.

II. SMEAR COUNT

The conversion factors for cts/min-100 cm^2 to dis/min-100 cm^2 for smear counts are given below:

A. Conversion Equation (Alpha)

$$\frac{\text{cts/min - (Bkgd)}}{\text{g x bf x sa x waf}} = \text{dis/min } \alpha$$

A geometry (g) of 0.43 is standard for all flat-plate counting using the Mylar spun top.

A backscatter factor (bf) of 1.0 was used when determining alpha activity on a filter media.

The self-absorption factor (sa) was assumed to be 1, unless otherwise determined.

If the energies of the isotope were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were not known, the (waf) of ^{239}Pu (0.713) was used.

The (waf) for alpha from ^{226}Ra plus daughters is 0.55.

B. Conversion Equation (Beta)

$$\frac{\text{cts/min - \{}\beta\text{ Bkgd (cts/min) + }\alpha\text{ cts/min\}}}{\text{g x bf x sa x waf}} = \text{dis/min } \beta$$

A geometry (g) of 0.43 is standard for all flat-plate counting using the Mylar spun top.

A backscatter factor (bf) of 1.1 was used when determining beta activity on a filter media.

A self-absorption factor (sa) was assumed to be 1, unless otherwise determined.

If the energies of the isotopes were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the (waf) of ^{90}Sr-^{90}Y (0.85) was used.

The (waf) for betas from ^{226}Ra plus daughters is 0.85.

APPENDIX 3

RADON-DETERMINATION CALCULATIONS

Calculations for determining radon concentrations in air samples collected with an Argonne National Laboratory-designed air sampler using HV-70 or LB5211 filter media are summarized in this appendix; the basic assumptions and calculations used to derive the air concentrations also are included.

I. RADON CONCENTRATIONS

The following postulates were assumed in deriving the radon (^{222}Rn) concentrations based on the RaC' alpha count results.

A. RaA, RaB, RaC, and RaC' are in equilibrium.

B. RaA is present only in the first count and not the 100-minute decay count.

C. One-half of the radon progeny is not adhered to airborne particulates (i.e., unattached fraction) and, therefore, is not collected on the filter media.

D. The geometry factor (g) is 0.43 for both the alpha and beta activity.

E. The backscatter factor (bf) for the alpha activity is 1.0.

F. The sample absorption factor (sa) for RaC' is 0.77.

G. The window air factor (waf) for RaC' is 0.8.

H. RaB and RaC, being beta emitters, are not counted in the alpha mode.

I. The half-life of the radon progeny is approximately 36 minutes, based on the combined RaB and RaC half-lives.

J. Thoron and long-lived alpha emitters are accounted for using the 360-minute decay count and the seven-day count, respectively.

K. For all practical purposes, RaC' decays at the rate of the composite of RaB and RaC, which is about 36 minutes.

The following postulates were assumed in deriving the thoron (^{220}Rn) concentrations.

L. ThA, ThB, ThC and ThC' are in equilibrium.

M. ThA and RaC' have decayed by the 360-minute decay count.

APPENDIX 3
(cont'd.)

N. The geometry factor (g), backscatter factor (bf), sample absorption factor (sa) and window air factor (waf) all are the same for thoron as for radon.

O. ThB and 64% of ThC, being beta emitters, are not counted in the alpha mode.

P. The half-life of the thoron progeny is 10.64 hours (638.4 minutes) based on the ThB half-life.

Q. For all practical purposes 36% of the ThC (alpha branch) and the ThC' decay at the same rate as ThB, which is 638.4 minutes.

R. The counter does not differentiate between the ThC alphas and the ThC' alphas.

The following postulates were assumed in deriving the actinon (^{219}Rn) concentrations.

S. AcA, AcB and AcC are in equilibrium.

T. AcA has decayed by the 100-minute decay count.

U. The geometry (g), backscatter (bf), sample absorption (sa) and window air factor (waf) factors all are the same for actinon as for radon.

V. AcB, being a beta emitter, is not counted in the alpha mode.

W. The half-life of the actinon progeny is 36.1 minutes based on the AcB half-life.

X. For all practical purposes, the AcC decays at the same rate as AcB, which is 36.1 minutes.

Y. 84% of the AcC decays by 6.62 MeV α emissions and 16% decays by 6.28 MeV α emissions.

The following postulate was assumed in deriving the long-lived concentration.

Z. The long-lived activity, as determined from the seven-day count, is assumed to be constant during the entire counting periods. This assumption is valid for isotopes with half-lives longer than a few years.

APPENDIX 3
(cont'd.)

II. EQUATIONS USED TO DERIVE AIR CONCENTRATIONS

$$A_o = \frac{A}{e^{-\lambda t}}$$

Where: A_o = activity (dis/min) present at the end of the sampling period (usually 40 minutes)

A = activity (dis/min) at some time, t, after end of the sampling period

t = time interval (minutes) from end of sampling period to counting interval (usually $\simeq$ 100 minutes)

$$\lambda = \frac{0.693}{t_{\frac{1}{2}}}$$

$t_{\frac{1}{2}}$ = half-life of isotope (min).

Concentration is determined by the equation:

$$C = \frac{A_o \lambda}{f} \times \frac{1}{1-e^{-\lambda t_s}}$$

Where: C = concentration (dis/min-m^3)

A_o = activity on filter media at end of sampling period (dis/min)

f = sampling rate (m^3/min = m^3/h x 1 h/60 minutes)

t_s = length of sampling time (minutes)

$$\lambda = \frac{0.693}{t_{\frac{1}{2}}}$$

$t_{\frac{1}{2}}$ = half-life of isotope or controlling parent (minutes).

III. ACTINON CORRECTION

Since the actinon (^{219}Rn) progeny (AcA, AcB & AcC) decays at the AcB half-life of 36 minutes, it cannot be distinguished from the radon (^{222}Rn) progeny using standard air sampling with HV-70 or LB5211 filter media and standard alpha-counting techniques. A positive displacement pump is used to collect a sample on Millipore membrane (0.5 to 0.8 μm) filter media. The sample rate is

APPENDIX 3
(cont'd.)

approximately 20 liters/minute for a sampling time of at least 90 minutes. The center portion of the sample is removed and counted in an alpha spectrometer which exhibits the 6.62 MeV AcC alpha emissions and the 7.69 MeV RaC' alpha emissions. If these two peaks are observed in the spectrum, then the following calculations are performed:

$$B_j = \sum_{i=1}^{n} b_{ij}$$

Where: B_j = summation of the counts in n channels under peak j

b_{ij} = the number of counts in channel i of peak j

j = 1 for the 6.62 MeV peak of actinon; 2 for the 7.69 MeV peak of radon

n = total number of channels in the summation.

The fractions of the activity with a 36-minute half-life due to actinon and radon are then:

$$\text{Actinon} = \frac{B_1/0.84}{B_1/0.84+B_2}$$

$$\text{Radon} = \frac{B_2}{B_1/0.84+B_2}$$

where 1 refers to actinon progeny and 2 refers to radon progeny.

IV. EXAMPLE CALCULATION

Data have been created to correspond to values likely to occur if all possible types of contamination are present in the air of a room where a sample is collected. The application of the equations for determining all types of activity and their concentrations are given below.

Data	f = 40 m^3/60 min	t_s = 40 min
at	t = 100 min	A^s = 2000 dis/min
at	t = 360 min	A = 140 dis/min
at	t = 7 days	A = 5 dis/min

For long-lived activity:

$$A_o = A = 5 \text{ dis/min}$$

$$C(L) = A_o/fxt_s = \frac{5}{40/60x40} = 0.19 \text{ dis/min-m}^3.$$

APPENDIX 3
(cont'd.)

For thoron:

$$A_o = \frac{140-5}{\exp - \frac{0.693 \times 360}{638.4}} = 199.6 \text{ dis/min}$$

$$C(Tn) = \frac{199.6 \times \frac{0.693}{638.4}}{40/60} \times \frac{1}{1 - \exp - \frac{0.693 \times 40}{638.4}} = 7.6 \text{ dis/min-m}^3.$$

For radon (^{222}Rn) and actinon (^{219}Rn), activity due to thoron at t = 100 min:

$$A = \frac{135}{\exp - \frac{0.693 \times 260}{638.4}} = 179 \text{ dis/min.}$$

Activity due to the isotopes with a 36 minute half-life:

$$A = 2000 - 179 - 5 = 1816 \text{ dis/min}$$

$$A_o = \frac{1816}{\exp - \frac{0.693 \times 100}{36}} = 12,449 \text{ dis/min}$$

$$C(36) = \frac{12,454 \times \frac{0.693}{36}}{40/60} \times \frac{1}{1 - \exp \frac{-0.693 \times 40}{36}} = 669.7 \text{ dis/min-m}^3.$$

When an actinon peak is seen at 6.62 MeV, the counts under the two peaks are summed. For example, if 10 channels are summed, the following counts are found:

For 6.62 MeV peak: 44 in 10 channels, where the 6.62 alpha emissions are 84% of the total.

For 7.69 MeV peak: 601 counts in 10 channels, where the 7.69 MeV alpha emissions are 100% of the total.

APPENDIX 3
(cont'd.)

B_1 = 44

$B_1/0.84$ = 52 counts

B_2 = 601 counts

Actinon = 52/653 = 0.08

Radon = 601/653 = 0.92

C(Rn) = C(36) x Radon% = 669.7 x 0.92 = 616.1 dis/min-m^3

C = C(36) x Actinon% = 669.7 x 0.08 = 53.6 dis/min-m^3.

Since we assume that on the average half of the progeny is not adhered to the airborne particulates, the above concentrations are then doubled to determine actual concentrations. We assume that there is no unattached fraction for the long-lived activity.

C actual = C measured x progeny correction factor

C(L) = 0.19 dis/min-m^3

C(Tn) = 7.6 dis/min-m^3 x 2 = 15.2 dis/min-m^3

C(An) = 53.6 dis/min-m^3 x 2 = 107.2 dis/min-m^3

C(Rn) = 616 dis/min-m^3 x 2 = 1232 dis/min-m^3.

These would then be the resulting concentrations in dis/min-m^3. To convert to pCi/ℓ, divide the concentrations by 2.2 x 10^3:

$$C(L) = \frac{0.19 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 8.6 \times 10^{-5} \text{ pCi}/\ell$$

$$C(Tn) = \frac{15.2 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 0.0068 \text{ pCi}/\ell$$

$$C(An) = \frac{107.2 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 0.048 \text{ pCi}/\ell$$

$$C(Rn) = \frac{1232 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 0.55 \text{ pCi}/\ell.$$

APPENDIX 4

SAMPLE PREPARATION AND ANALYSIS GENERIC PROTOCOL

I. SOIL-SAMPLE PREPARATION

Soil samples are acquired as previously described. These samples are bagged and identified at the collection site and returned to ANL. If there is an indication of radioactive contamination, the sample is sealed in a Nalgene jar. At ANL, the soil samples are logged into the soil-sample book, and each sample is weighed (on a tared balance scale) and the weight is marked on the container. This weight is recorded in the soil book as a "net weight."

After all samples are marked, weighed, and recorded, they are dried. Each sample is placed in a pyrex beaker marked with the sample identification number. If more than one beaker is necessary, additional numbers (e.g., 1-3, 2-3, 3-3) are used. The original containers are saved for repackaging the dried samples. The beaker is set in an 80°C oven until the soil is dry (approximatley 48 hours). The sample is returned to the original container and reweighed using a tared balance scale. This weight is also marked on the container and in the soil-sample book, where it is referred to as a "dry weight."

After all the samples are returned to their original containers, the milling process is started. Each dried sample is transferred to a 2.3-gallon ceramic mill jar containing mill balls (1¼" x 1¼" Burundum cylinders). The mill jar number is marked on the original container. The jars are sealed and the samples are milled for two hours or until sufficient material is produced to obtain 100 g and 5 g samples for analyses. The samples are milled six at a time. A second set of six jars is prepared while the milling of the first set is proceeding. After each sample is milled, the mill balls are removed with tongs and placed in a tray. A large plastic bag is inverted over the mill jar. Both are inverted and shaken until all the soil is transferred to the bag. If the soil plates the inside of the mill jar, a small paint brush is used to loosen the soil before the jar is inverted. A separate brush is used for each jar to prevent cross-contamination of the soil samples.

After milling, each sample is sieved through a number 30 standard testing sieve (600 μ mesh) and transferred to a 12" x 12" ziplock bag. Rocks and dross are bagged separately. The bags are marked with the sample number, the sieve number and R(rocks) or S(soil). The balance is tared and the weights of the soil (or rocks) are measured and recorded in the soil-sample book. A 100-g sample of the sieved material is transferred to a 4-oz. Nalgene bottle. These samples are analyzed by suitable analytical techniques, including, as a minimum, gamma spectroscopy (GeLi). A 5-g sample of the sieved material is transferred to a 1-oz Nalgene bottle. One gram of this sample is used for the determination of uranium by laser fluorometry; 100 g are needed for radiochemical analysis of Pu, Am, and Th if these analyses are required. The bottles containing these weighed samples are marked with sample number and date and this information is recorded in the soil-sample book. The rocks (and dross) and remaining soil are placed in storage.

APPENDIX 4
(cont'd.)

The sieves, mill jars, and Burundum milling balls used in this work are classified in two sets. One set is used for background samples exclusively. The other set is used for all samples from suspect areas. Soil samples with elevated levels of radioactivity based on instrument measurements are milled in one-gallon Nalgene bottles using Burundum balls from the set used for suspect samples. After use, these balls are either decontaminated (see below) or disposed of as radioactive waste. The Nalgene bottles are always disposed of as radioactive waste. The sieves used for these samples are also from the set used for suspect samples and are decontaminated after use.

II. EQUIPMENT DECONTAMINATION

The care of the milling apparatus is as important as the actual sample preparation. Proper care prevents cross-contamination of successive samples. The beakers used to dry the samples are washed thoroughly by placing a small amount of Haemo-Sol in each beaker and filling with warm water. The beaker is then scrubbed thoroughly on the inside and scoured on the outside with scouring powder. The beakers are rinsed three times with tap water and three times with demineralized water, and finally dried thoroughly before reuse.

The milling apparatus (tongs, brushes, milling jars, lids and milling balls) are rinsed. The tongs and brushes are washed thoroughly with Haemo-Sol. Eight Burundum balls are returned to each milling jar along with about one pint of clean road gravel, one spoon of Haemo-Sol, one spoon of scouring powder with bleach, and one quart of water. The lid is tightened on the jar and the jar is placed on the rolling mill and rolled for approximately two hours or until the balls and the inside of the jar appear to be physically clean. After this time, the mill jar is removed from the rolling mill and its contents are dumped into a screen or basket. The lid and balls are then rinsed thoroughly three times with tap water followed by three times with demineralized water. The inside of the jar is rinsed until it is absolutely clean. The milling apparatus is air-dried with warm air until absolutely dry. Room air is drawn through Mill jars with a hose which is attached to a fume hood or specially constructed drying box.

The sieves are rinsed, washed in Haemo-Sol, thoroughly rinsed (three times with tap water, followed by three rinses with demineralized water) and then air dried as above before reuse.

III. WATER AND SLUDGE

Water samples are collected in 0.1-liter, 0.5-liter and/or 1-liter quantities as deemed appropriate. These samples are forwarded directly to a certified radiochemistry laboratory for preparation and analysis. The customary analysis procedure consists of filtration to obtain the suspended solids followed by evaporation to obtain the dissolved solids. Both suspended and dissolved solids are analyzed by appropriate radiochemical analytical techniques.

APPENDIX 4
(cont'd.)

Sludge samples are collected in 0.1-liter bott¹ s and are processed as outlined above for water samples.

IV. VEGETATION, TRASH AND RUBBLE

Samples of potentially contaminated vegetation, trash (e.g. piping, ducts, conduit, etc.) and rubble are collected, bagged, and labeled at the site and returned to ANL for analysis.

Vegetation samples are initially weighed and transferred to Marinelli beakers for gamma spectrometric analysis. Then they are ashed, reweighed, and analyzed by appropriate analytical techniques.

Trash and rubble samples are forwarded to a certified radiochemistry laboratory for analysis.

V. TRITIUM FROM SOLID MATERIALS

Samples of solid materials (e.g., concrete) suspected of containing tritium are collected, broken into small pieces, and submitted to a certified radiochemistry laboratory for analysis. The standard analytical procedure consists of transferring a 20-40 g sample to a ceramic boat followed by heating in a tube furnace at 425°C for a period of two hours (~ 40 min to reach temperature and ~ 80 min heating at temperature). Helium is used as a flow gas through the tube during heating, and the tritium is collected in two traps on the downstream side of the furance. The first trap is immersed in an ordinary ice bath (0°C); the second trap is immersed in a CO_2-Freon bath (-57°C). The collected tritiated water from both traps is combined, made up to a known volume, and an aliquot taken for liquid scintillation counting of the tritium.

VI. ANALYSIS PROCEDURES

A 100-g fraction from each soil sample is analyzed by high resolution gamma-ray spectroscopy using a germanium crystal detector coupled to a multichannel analyzer. This analysis allows for a quantitative determination of the ^{226}Ra decay chain (via the 609 keV γ-ray of ^{214}Bi) and the ^{232}Th decay chain (via the 911 keV γ-ray of ^{228}Ac, as well as any other gamma emitting radionuclide, e.g., ^{137}Cs) present in the soil.

The total uranium (elemental) present in the soil is determined by an acid leach of the soil sample followed by laser fluorometry of the leached sample.

Thorium analysis consists of an acid leach of the soil (using a ^{234}Th spike for yield determination) followed by plating a thin source of the radiochemically separated thorium and determining the thorium isotopes (^{228}Th and ^{232}Th) by alpha spectroscopy.

APPENDIX 4
(cont'd.)

The results of the above measurements allow for quantitative determination of the relative amounts of normal uranium, natural uranium, tailings (i.e., ^{226}Ra decay chain), thorium (^{232}Th), mesothorium (^{228}Ra decay chain), and thorium (^{228}Th) decay chain present in the contaminated material.

A mass spectrometric analysis of the uranium fraction is conducted when it is known or is is surmised that depleted or enriched uranium might be present.

APPENDIX 5

CALCULATION OF NORMAL-URANIUM SPECIFIC ACTIVITY

The specific activity for normal uranium was obtained by summing the measured specific activities for the individual isotopes weighted according to their normal abundances. Best values for these specific activities of ^{235}U and ^{238}U were taken from A. H. Jaffey et al., Phys. Rev. C 4 1889 (1971). The half-life for each isotope was taken from David C. Kocher, "Radioactive Decay Tables-A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments" (1981). The percent abundances were taken from N. E. Holden, BNL-NCD-50605 (1977). Atomic weights were taken from the "Handbook of Chemistry and Physics," 52nd Edition (1971). The specific activity of ^{234}U was calculated from the half-life.

Isotope	Specific Activity	Half-life (years)	Abundance (atom %)	Atomic Weight (grams)	Abundance (wt %)
^{234}U	1.387×10^4 dis/min-μg	2.445×10^5	0.0054	234.0409	0.0053
^{235}U	4.798 dis/min-μg	7.038×10^8	0.720	235.0439	0.7110
^{238}U	0.746 dis/min-μg	4.4683×10^9	99.2746	238.0508	99.2837
			100.0000		100.0000

where $(\text{wt }\%)_i =$

$$\frac{(\text{atom }\%)_i \ (\text{atomic weight})_i}{\sum_j (\text{atom }\%)_j \ (\text{atomic weight})_j} = \frac{(\text{atom }\%)_i \ (\text{atomic weight})_i}{238.0289}$$

Specific activity for normal uranium:

$$0.746 \times 0.99284 \times 2 = 1.481 \quad \text{dis/min-μg from } ^{234} \text{ \& } ^{238}U$$
$$4.798 \times 0.00711 = 0.034 \quad \text{dis/min-μg from } ^{235}U$$
$$1.515 \quad \text{dis/min-μg for normal U}$$

or $(1.515 \text{ dis/min-μg})/(2.22 \text{ dis/min-pCi}) = 0.683 \text{ pCi/μg}$

where ^{234}U is assumed to be in secular equilibrium with the ^{238}U parent.

Note that 2.25% of the total activity is due to ^{235}U and 48.87% each is due to ^{234}U and ^{238}U.

APPENDIX 6

PERTINENT RADIOLOGICAL REGULATIONS, STANDARDS, AND GUIDELINES

Excerpts From

I. DRAFT AMERICAN NATIONAL STANDARD

N13.12

Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use

Where potentially contaminated surfaces are not accessible for measurement (as in some pipes, drains, and ductwork), such property shall not be released pursuant to this standard, but shall be made the subject of case-by-case evaluation.

Property shall not be released for uncontrolled use unless measurements show the total and removable contamination levels to be no greater than the values in Table 1 or Table 2. (The values in Table 2 are easier to apply when the contaminants cannot be individually identified.)

Coatings used to cover the contamination shall not be considered a solution to the contamination problem. That is, the monitoring techniques shall be sufficient to determine, and such determination shall be made, that the total amount of contamination present on and under any coating does not exceed the Table 1 or Table 2 values before release.

APPENDIX 6
(Cont'd.)

TABLE 1

SURFACE CONTAMINATION LIMITS*

Contaminants			Limit (Activity) (dis/min-100 cm^2)+	
Group	Description	Nuclides (Note 1)	Removable	Total (Fixed plus Removable)
1	Nuclides for which the nonoccupational MPC$_a$ (Note 2) is 2×10^{-13} $\mu Ci/m^3$ or less or for which the nonoccupational MPC$_w$ (Note 4) is 2×10^{-7} $\mu Ci/m^3$ or less	^{227}Ac $^{241,242m,243}Am$ $^{249,250,251,252}Cf$ $^{243,244,245,246,247,248}Cm$ $^{125,129}I$ ^{237}Np ^{231}Pa ^{210}Pb $^{238,239,240,242,244}Pu$ $^{226,228}Ra$ $^{228,230}Th$	20	Nondetectable (Note 3)
2	Those nuclides not in Group 1 for which the nonoccupational MPC$_a$ (Note 2) is 1×10^{-12} $\mu Ci/m^3$ or less for which the nonoccupational MPC$_w$ (Note 4) is 1×10^{-6} $\mu Ci/m^3$ or less	^{254}Es ^{256}Fm $^{126,131,133}I$ ^{210}Po ^{223}Ra ^{90}Sr ^{232}Th ^{232}U	200	2000 α Nondetectable β,γ (Note 5)
3	Those nuclides not in Group 1 or Group 2		1000	5000

APPENDIX 6
(Cont'd.)

SURFACE CONTAMINATION LIMITS

*The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any square meter of surface shall be considered to be contaminated above the limit L, applicable to 100 cm^2, if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i}^{n} S_i \geq L$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3 L.

+Disintegrations per minute per square decimeter.

NOTES:

(1) Values presented here are obtained from the Code of Federal Regulations, Title 10, Part 20, April 30, 1975. The most limiting of all given MPC values (for example, soluble versus insoluble) are to be used. In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fraction shall be less than 1.

(2) Maximum permissible concentration in air applicable to continuous exposure of members of the public as published by or derived from an authoritative source such as the National Committee on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), or the Nuclear Regulatory Commission (NRC). From the Code of Federal Regulations, Title 10, Part 20, Appendix B, Table 2, Column 1.

(3) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group 1 contaminants uniformly spread over 100 cm^2.

(4) Maximum permissible concentration in water applicable to members of the public.

(5) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group 2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey for unconditional release should be performed in areas where the background is ≤ 100 counts per minute. When the survey must be performed in a background exceeding 100 counts per minute, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

APPENDIX 6
(Cont'd.)

TABLE 2
ALTERNATE SURFACE CONTAMINATION LIMITS

(All Alpha Emitters, except U_{nat} and Th_{nat}, Considered as a Group)*

Contamination Contingencies	Limit (Activity) (dis/min-100 cm^2)+ Removable	Total (Fixed Plus Removable
If the contaminant cannot be identified; or if alpha emitters other than U_{nat} (Note 1) and Th_{nat} are present; or if the beta emitters comprise ^{227}Ac or ^{228}Ra.	20	Nondetectable (Note 2)
If it is known that all alpha emitters are generated from U_{nat} (Note 1) and Th_{nat}; and if beta emitters are present that, while not identified, do not include ^{227}Ac, ^{125}I, ^{226}Ra, and ^{228}Ra.	200	2000 α Nondetectable β,γ (Note 3)
If it is known that alpha emitters are generated only from U_{nat} (Note 1) and Th_{nat} in equilibrium with its decay products; and if the beta emitters, while not identified, do not include ^{227}Ac, ^{125}I, ^{129}I, ^{90}Sr, ^{223}Ra, ^{228}Ra, ^{126}I, ^{131}I and ^{133}I.	1000	5000

APPENDIX 6
(Cont'd.)

ALTERNATE SURFACE CONTAMINATION LIMITS

*The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any square meter of surface shall be considered to be contaminated above the limit L, applicable to 100 cm^2, if (1) from measurements of a representative number n of sections it is determined that $1/n \Sigma_n S_i \geq L$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3 L.

+Disintegrations per minute per square decimeter.

NOTES:

(1) U_{nat} and decay products.

(2) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group 1 contaminants uniformly spread over 100 cm^2.

(3) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group 2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey of unconditional release should be performed in areas where the background is ≤ 100 counts per minute. When the survey must be performed in a background exceeding 100 counts per minute, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

APPENDIX 6
(Cont'd.)

II. U.S.NUCLEAR REGULATORY COMMISSION
DIVISION OF FUEL CYCLE AND MATERIAL SAFETY
WASHINGTON, D.C.
July 1982

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BY-PRODUCT SOURCE, OR SPECIAL NUCLEAR MATERIAL

(These have been retyped for purposes of this report)

The instructions in this guide, in conjunction with Table 1, specify the radioactivity and radiation exposure rate limits which should be used in accomplishing the decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control will be considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 1 prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or duct work shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or duct work. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such request must:

APPENDIX 6
(cont'd.)

a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

a. Identify the premises.

b. Show that reasonable effort has been made to eliminate residual contamination.

c. Describe the scope of the survey and general procedures followed.

d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

APPENDIX 6
(Cont'd.)

TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LIMITS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, ^{235}U, ^{238}U and associated decay products	5000 dis/min-100 cm^2 α	15,000 dis/min-100 cm^2 α	1000 dis/min-100 cm^2 α
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	100 dis/min-100 cm^2	300 dis/min-100 cm^2	20 dis/min-100 cm^2
Th-nat, ^{232}Th ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	1000 dis/min-100 cm^2	3,000 dis/min-100 cm^2	200 dis/min-100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above.	5000 dis/min-100 cm^2 $\beta\gamma$	15,000 dis/min-100 cm^2 $\beta\gamma$	1000 dis/min-100 cm^2 $\beta\gamma$

APPENDIX 6
(Cont'd.)

TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LEVELS

[a]Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpba and beta-gamma emitting nuclides should apply independently.

[b]As used in this table, dis/min (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

APPENDIX 6
(Cont'd.)

III. SURGEON GENERAL'S GUIDELINES
as included in 10 CFR Part 712
Grand Junction Remedial Action Criteria

712.1 Purpose

(a) determination by DOE of the need for, priority of and selection of appropriate remedial action to limit the exposure of individuals in the area of Grand Junction, Colorado, to radiation emanating from uranium mill tailings which have been used as construction-related material.

(b) The regulations in this part are issued pursuant to Pub. L. 92-314 (86 Stat. 222) of June 16, 1972.

712.2 Scope

The regulations in this part apply to all structures in the area of Grand Junction, Colorado, under or adjacent to which uranium mill tailings have been used as a construction-related material between January 1, 1951, and June 16, 1972, inclusive.

712.3 Definitions

As used in this part:

(a) "Administrator" means the Administrator of Energy Research and Development or his duly authorized representative.

(b) "Area of Grand Junction, Colorado," means Mesa County, Colorado.

(c) "Background" means radiation arising from cosmic rays and radioactive material other than uranium mill tailings.

(d) "DOE" means the U.S. Department of Energy or any duly authorized representative thereof.

(e) "Construction-related material" means any material used in the construction of a structure.

(f) "External gamma radiation level" means the average gamma radiation exposure rate for the habitable area of a structure as measured near floor level.

APPENDIX 6
(Cont'd.)

(g) "Indoor radon daughter concentration level" means that concentration of radon daughters determined by: (1) averaging the results of six air samples each of at least 100 hours duration, and taken at a minimum of 4-week intervals throughout the year in a habitable area of a structure, or (2) utilizing some other procedure approved by the Commission.

(h) "Milliroentgen" (mR) means a unit equal to one-thousandth (1/1000) of a roentgen which roentgen is defined as an exposure dose of X or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying one electrostatic unit of quantity of electricity of either sign.

(i) "Radiation" means the electromagnetic energy (gamma) and the particulate radiation (alpha and beta) which emanate from the radioactive decay of radium and its daughter products.

(j) "Radon daughters" means the consecutive decay products of radon-222. Generally, these include Radium A (polonium-218), Radium B (lead-214), Radium C (bismuth-214), and Radium C' (polonium-214).

(k) "Remedial action" means any action taken with a reasonable expectation of reducing the radiation exposure resulting from uranium mill tailings which have been used as construction-related material in and around structures in the area of Grand Junction, Colorado.

(l) "Surgeon General's Guidelines" means radiation guidelines related to uranium mill tailings prepared and released by the Office of the U.S. Surgeon General, Department of Health, Education and Welfare on July 27, 1970.

(m) "Uranium mill tailings" means tailings from a uranium milling operation involved in the Federal uranium procurement program.

(n) "Working Level" (WL) means any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

712.4 Interpretations

Except as specifically authorized by the Administrator in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of DOE other than a written interpretation by the General Counsel will be recognized to be binding upon DOE.

712.5 Communications

Except where otherwise specified in this part, all communications concerning the regulations in this part should be addressed to the Director, Division of Safety, Standards, and Compliance, U.S. Department of Energy, Washington, D.C. 20545.

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712.6 General radiation exposure level criteria for remedial action.

The basis for undertaking remedial action shall be the applicable guidelines published by the Surgeon General of the United States. These guidelines recommended the following graded action levels for remedial action in terms of external gamma radiation level (EGR) and indoor radon daughter concentration level (RDC) above background found within dwellings constructed on or with uranium mill tailings.

EGR	RDC	Recommendation
Greater than 0.1 mR/h	Greater than 0.05 WL	Remedial action indicated.
From 0.05 to 0.1 mR/h	From 0.01 to 0.05 WL	Remedial action may be suggested.
Less than 0.05 mR/h	Less than 0.01 WL	No remedial action indicated

712.7 Criteria for determination of possible need for remedial action

Once it is determined that a possible need for remedial action exists, the record owner of a structure shall be notified of that structure's eligibility for an engineering assessment to confirm the need for remedial action and to ascertain the most appropriate remedial measure, if any. A determination of possible need will be made if as a result of the presence of uranium mill tailings under or adjacent to the structure, one of the following criteria is met:

(a) Where DOE approved data on indoor radon daughter concentration levels are available

(1) For dwellings and schoolrooms: An indoor radon daughter concentration level of 0.01 WL or greater above background.

(2) For other structures: An indoor radon daughter concentration level of 0.03 WL or greater above background.

(b) Where DOE approved data on indoor radon daughter concentration levels are not available:

(1) For dwellings and schoolrooms:

(i) An external gamma radiation level of 0.05 mR/h or greater above background.

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(ii) An indoor radon daughter concentration level of 0.01 WL or greater above background (presumed).

(A) It may be presumed that if the external gamma radiation level is equal to or exceed 0.02 mR/h above background, the indoor radon daughter concentration level equals or exceeds 0.01 WL above background.

(B) It should be presumed that if the external gamma radiation level is less than 0.001 mR/h above background, the indoor radon daughter concentration level is less than 0.01 WL above background, and no possible need for remedial actions exists.

(C) If the external gamma radiation level is equal to or greater than 0.001 mR/h above background but is less than 0.02 mR/h above background, measurements will be required to ascertain the indoor radon daughter concentration level.

(2) For other structures:

(i) An external gamma radiation level of 0.15 mR/h above background averaged on a room-by-room basis.

(ii) No presumptions shall be made on the external gamma radiation level/indoor radon daughter concentration level relationship. Decisions will be made in individual cases based upon the results of actual measurements.

712.8 Determination of possible need for remedial action where criteria have not been met.

The possible need for remedial action may be determined where the criteria in 712.7 have not been met if various other factors are present. Such factors include, but are not necessarily limited to, size of the affected area, distribution of radiation levels in the affected area, amount of tailings, age of individuals occupying affected area, occupancy time, and use of the affected area.

712.9 Factors to be considered in determination of order of priority for remedial action.

In determining the order or priority for execution of remedial action, consideration shall be given, but not necessarily limited to, the following factors:

(a) Classification of structure. Dwellings and schools shall be considered first.

(b) Availability of data. Those structures for which data on indoor radon daughter concentration levels and/or external gamma radiation levels are available when the program starts and which meet the criteria in 712.7 will be considered first.

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(c) Order of application. Insofar as feasible remedial action will be taken in the order in which the application is received.

(d) Magnitude of radiation level. In general, those structures with the highest radiation levels will be given primary consideration.

(e) Geographical location of structures. A group of structures located in the same immediate geographical vicinity may be given priority consideration particularly where they involve similar remedial efforts.

(f) Availability of structures. An attempt will be made to schedule remedial action during those periods when remedial action can be taken with minimum interference.

(g) Climatic conditions. Climatic conditions or other seasonable considerations may affect the scheduling of certain remedial measures.

712.10 Selection of appropriate remedial action.

(a) Tailings will be removed from those structures where the appropriately averaged external gamma radiation level is equal to or greater than 0.05 mR/h above background in the case of dwellings and schools and 0.15 mR/h above background in the case of other structures.

(b) Where the criterion in paragraph (a) of this section is not met, other remedial action techniques, including but not limited to sealants, ventilation, and shielding may be considered in addition to that of tailings removal. DOE shall select the remedial action technique or combination of techniques, which it determines to be the most appropriate under the circumstances.

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IV. 40 CFR Part 192
HEALTH AND ENVIRONMENTAL PROTECTION STANDARDS
FOR
URANIUM MILL TAILINGS

SUBPART A--Standards for the Control of Residual Radioactive Materials from Inactive Uranium Processing Sites

192.00 Applicability

This subpart applies to the control of residual radioactive material at designated processing or depository sites under Section 108 of the Uranium Mill Tailings Radiation Control Act of 1978 (henceforth designated "the Act"), and to restoration of such sites following any use of subsurface minerals under Section 104(h) of the Act.

192.01 Definitions

(a) Unless otherwise indicated in this subpart, all terms shall have the same meaning as in Title I of the Act.

(b) Remedial action means any action performed under Section 108 of the Act.

(c) Control means any remedial action intended to stabilize, inhibit future use of, or reduce emissions or effluents from residual radioactive materials.

(d) Disposal site means the region within the smallest perimeter of residual radioactive material (excluding cover materials) following completion of control activities.

(e) Depository site means a disposal site (other than a processing site) selected under Section 104(b) or 105(b) of the Act.

(f) Curie (Ci) means the amount of radioactive material that produces 37 billion nuclear transformation per second. One picocurie (pCi) = 10^{-12} Ci.

192.02 Standards

Control shall be designed* to:

(a) be effective for up to one thousand years, to the extent reasonably achievable, and, in any case, for at least 200 years, and,

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*Because the standard applies to design, monitoring after disposal is not required to demonstrate compliance.

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(b) provide reasonable assurance that releases of radon-222 from residual radioactive material to the atmosphere will not:

(1) exceed an average** release rate of 20 picocuries per square meter per second, or

(2) increase the annual average concentration of radon-222 in air at or above any location outside the disposal site by more than one-half picocurie per liter.

SUBPART B--Standards for Cleanup of Open Lands and Buildings Contaminated with Residual Radioactive Materials from Inactive Uranium Processing Sites

192.10 Applicability

This subpart applies to land and buildings which are part of any processing site designated by the Secretary of Energy under Pub. L. 95-604, Section 102. Section 101 of Pub. L. 95-604, states that "processing site" means--

(a) any site, including the mill, containing residual radioactive materials at which all or substantially all of the uranium was produced for sale to any Federal agency prior to January 1, 1971, under a contract with any Federal agency, except in the case of a site at or near Slick Rock, Colorado, unless--

(1) such site was owned or controlled as of January 1, 1978, or is thereafter owned or controlled, by an Federal agency, or

(2) a license [issues by the (Nuclear Regulatory) Commission or its predecessor agency under the Atomic Energy Act of 1954 or by a State as permitted under Section 274 of such Act] for the production at such site of any uranium or thorium product dervied from ores is in effect on January 1, 1978, or is issued or renewed after such date; and

(b) Any other real property or improvement thereon which--

(1) is in the vicinity of such site, and

(2) is determined by the Secretary, in consultation with the Commission, to be contaminated with residual radioactive materials derived from such site.

**This average shall apply over the entire surface of the disposal site and over at least a one-year period. Radon will come from both residual radioactive materials and from materials covering them. Radon emissions from the covering materials should be estimated as part of developing a remedial action plan for each site. The standard, however, applies only to emissions from residual radioactive materials to the atmosphere.

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192.11 Definitions

(a) Unless otherwise indicated in this subpart, all terms shall have the same meaning as defined in Title I of the Act or in Subpart A.

(b) Land means any surface or subsurface land that is not part of a disposal site and is not covered by an occupiable building.

(c) Working Level (WL) means combination of short-lived radon decay products in one liter of air that will result in the ultimate emission of alpha particles with a total energy of 130 billion electron volts.

(d) Soil means all unconsolidated materials normally found on or near the surface of the earth including, but not limited to silts, clays, sands, gravel, and small rocks.

192.12 Standards

Remedial actions shall be conducted so as to provide reasonable assurance that, as a result of residual radioactive materials from any designated processing site:

(a) the concentration of radium-226 in land averaged over any area of 100 square meters shall not exceed the background level by more than---

(1) 5 pCi/g, averaged over the first 15 cm of soil below the surface, and

(2) 15 pCi/g, averaged 15 cm thick layers of soil more than 15 cm below the surface.

(b) in any occupied or habitable building---

(1) the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL. In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL, and

(2) the level of gamma radiation shall not exceed the background level by more than 20 microroentgens per hour.

SUBPART C--Implementation

192.20 Guidance for Implementation

Section 108 of the Act requires the Secretary of Energy to select and perform remedial actions with the concurrence of the Nuclear Regulatory Commission and the full participation of any State that pays part of the cost, and in consultation, as appropriate, with affected Indian Tribes and the Secretary of the Interior. These parties, in their respective roles under Section 108, are referred to hereafter as "the implementing agencies."

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The implementing agencies shall establish methods and procedures to provide "reasonable assurance" that the provisions of Subparts A and B are satisfied. This should be done as appropriate through use of analytic models and site-specific analyses, in the case of Subpart A, and for Subpart B through measurements performed within the accuracy of currently available types of field and laboratory instruments in conjunction with reasonable survey and sampling procedures. These methods and procedures may be varied to suit conditions at specific sites. In particular:

(a) The purpose of Subpart A is to provide for long-term stabilization and isolation in order to inhibit misuse and spreading of residual radioactive materials, control releases of radon to air, and protect water. Subpart A may be implemented through analysis of the physical properties of the site and the control system and projection of the effects of natural processes over time. Events and processes that could significantly affect the average radon release rate from the entire disposal site should be considered. Phenomena that are localized or temporary, such as local cracking or burrowing of rodents, need to be taken into account only if their cumulative effect would be significant in determining compliance with the standard. Computational models, theories, and prevalent expert judgment may be used to decide that a control system design will satisfy the standard. The numerical range provided in the standard for the longevity of the affectiveness of the control of residual radioactive materials allows for consideration of the various factors affecting the longevity of control and stabilization methods and their costs. These factors have different levels of predictability and may vary for the different sites.

Protection of water should be considered in the analysis of reasonable assurance of compliance with the provisions of Section 192.02. Protection of water should be considered on a case-specific basis, drawing on hydrological and geochemical surveys and all other relevant data. The hydrologic and geologic assessment to be conducted at each site should include a monitoring program sufficient to establish background groundwater quality through one or more upgradient wells, and identify the presence and movement of plumes associated with the tailings piles.

If contaminants have been released from a tailings pile, an assessment of the location of the contaminants and the rate and direction of movement of contaminated groundwater, as well as its relative contamination, should be made. In addition, the assessment should identify the attenuative capacity of the unsaturated and saturated zone to determine the extent of plume movement. Judgments on the possible need for remedial or protective actions for groundwater aquifers should be guided by relevant considerations described in EPA's hazardous waste management system (47 FR 32274, July 26, 1982) and by relevant State and Federal Water Quality Criteria for anticipated or existing uses of water over the term of the stabilization. The decision on whether to institute remedial action, what specific action to take, and to what levels an aquifer should be protected or restored should be made on a case-by-case basis taking into account such factors as technical feasibility of improving the aquifer in its hydrogeologic setting, the cost of applicable restorative or protective programs, the present and future value of the aquifer as a water resource, the availability of alternative water supplies, and the degree to which human exposure is likely to occur.

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(b) Compliance with Subpart B, to the extent practical, should be demonstrated through radiation surveys. Such surveys may, if appropriate, be restricted to locations likely to contain residual radioactive materials. These surveys should be designed to provide for compliance averaged over limited areas rather than point-by-point compliance with the standards. In most cases, measurement of gamma radiation exposure rates above and below the land surface can be used to show compliance with Section 192.12(a). Protocols for making such measurements should be based on realistic radium distributions near the surface rather than extremes rarely encountered.

In Section 192.12(a), the "background level" refers to native radium concentration in soil. Since this may not be determinable in the presence of contamination by residual radioactive materials, a surrogate "background level" may be established by simple or indirect (e.g., gamma radiation) measurements performed nearby but outside of the contaminated location.

Compliance with Section 192.12(b) may be demonstrated by methods that the Department of Energy has approved for use under PL 92-314 (10 CFR 712), or by other methods that the implementing agencies determine are adequate. Residual radioactive materials should be removed from buildings exceeding 0.03 WL so that future replacement buildings will not pose a hazard [unless removal is not practical--see Section 192.21(c)]. However, sealants, filtration, and ventilation devices may provide reasonable assurance of reductions from 0.03 WL to below 0.02 WL. In unusual cases, indoor radiation may exceed the levels specified in Section 192.12(b) due to sources other than residual radioactive materials. Remedial actions are not required in order to comply with the standard when there is reasonable assurance that residual radioactive materials are not the cause of such an excess.

192.21 Critiera for Applying Supplemental Standards

The implementing agencies may (and in the case of Subsection (f) shall) apply standards under Section 192.22 in lieu of the standards of Subparts A or B if they determine that any of the following circumstances exists:

(a) Remedial actions required to satisfy Subparts A or B would pose a clear and present risk of injury to workers or to members of the public, notwithstanding reasonable measures to avoid or reduce risk.

(b) Remedial actions to satisfy the cleanup standards for land, Section 192.12(a), or the acquisition of minimum materials required for control to satisfy Section 192.02(b), would, notwithstanding reasonable measures to limit damage, directly produce environmental harm that is clearly excessive compared to the health benefits to persons living on or near the site, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that may reasonably be anticipated.

(c) The estimate cost of remedial action to satisfy Section 192.12(a) at a "vicinity" site [described under Section 101(6)(B) of the Act] is unreasonably high relative to the long-term benefits, and the residual radioactive materials

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do not pose a clear present or future hazard. The likelihood that buildings will be erected or that people will spend long periods of time at such a vicinity site should be considered in evaluating this hazard. Remedial action will generally not be necessary where residual radioactive materials have been placed semi-permanently in a location whre site-specific factors limit their hazard and from which they are costly or difficult to remove, or where only minor quantities of residual radioactive materials are involved. Examples are residual radioactive materials under hard surface public roads and sidewalks, around public sewer lines, or in fence post foundations. Supplemental standards should not be applied at such sites, however, if individuals are likely to be exposed for long periods of time to radiation from such materials at levels above those that would prevail under Section 192.12(a).

(d) The cost of a remedial action for cleanup of a building under Section 192.12(b) is clearly unreasonably high relative to the benefits. Factors that should be included in this judgment are the anticipated period of occupancy, the incremental radiation level that would be affected by the remedial action, the residual useful lifetime of the building, the potential for future construction at the site, and the applicability of less costly remedial methods than removal of residual radioactive materials.

(e) There is no known remedial action.

(f) Radionuclides other than radium-226 and its decay products are present in sufficient quantity and concentration to constitute a significant radiation hazard from residual radioactive materials.

192.22 Supplemental Standards

Federal agencies implementing Subparts A and B may in lieu thereof proceed pursuant to this section with respect to generic or individual situations meeting the eligibility requirements of Section 192.21.

(a) When one or more of the criteria of Section 192.21(a) through (e) applies, the implementing agencies shall select and perform remedial actions that come as close to meeting the otherwise applicable standard as is reasonable under the circumstances.

(b) When Section 192.21(f) applies, remedial actions shall, in addition to satisfying the standards of Subparts A and B, reduce other residual radioactivity to levels that are as low as is reasonably achievable.

(c) The implementing agencies may make general determinations concerning remedial actions under this Section that will apply to all locations with specified characteristics, or they may make a determination for a specific location. When remedial actions are proposed under this Section for a specific location, the Department of Energy shall inform any private owners and occupants of the affected location and solicit their comments. The Department of Energy shall provide any such comments to the other implementing agencies. The Department of Energy shall also periodically inform the Environmental Protection Agency of both general and individual determinations under the provisions of this section.

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V. EXCERPTS FROM LA-UR-79-1865-Rev.,

"Interim Soil Limits for D&D Projects"

Table XXIII. Recommended Soil Limits[a,b] (in pCi/g)

	Ingestion				
	Inhalation	Home Gardener	Full Diet	External Radiation	All Pathways[c]
^{231}Pa	50	740	150	250	40
^{227}Ac	200[d]	4,900	1,000	300	120[d]
^{232}Th	45	670	140	40	20
^{228}Th	1,000	37,000	7,800	55	50
^{230}Th (No Daught.)	300	4,400	940	36,000	280
^{238}U-^{234}U	750	44	8	6,000	40
^{90}Sr	$2x10^6$	100	19	-	100
^{137}Cs	$7x10^6$	800	1	90	80

[a] Soil limits for ^{241}Am and $^{239,240}Pu$ are available from EPA recommendations, and a soil limit for ^{226}Ra has been reported by Healy and Rodgers.

[b] Limits are to apply to only one nuclide present in the soil. If more than one is present, a weighted average should apply.

[c] Based on a diet of a home gardener.

[d] Modified from LA-UR-79-1865-Rev. values to correct error.

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VI. EXCERPTS FROM DOE 5480.1, Chapter XI

"Requirements for Radiation Protection"

Exposure of Individuals and Population Groups in Uncontrolled Areas. Exposures to members of the public shall be as low as reasonably achievable levels within the standards prescribed below.

Radiation Protection Standards
for Internal and External Exposure
of Members of the Public

	Annual Dose Equivalent or Dose Commitment	
Type of Exposure	Based on Dose to Individuals at Points of Maximum Probable Exposure	Based on Average Dose to a Suitable Sample of the Exposed Population
Whole body, gonads, or bone marrow	0.5 rem (or 500 mrem)	0.17 rem (or 170 mrem)
Other organs	1.5 rem (or 1500 mrem)	0.5 rem (or 500 mrem)

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CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

Element (atomic number)	Isotope	soluble (S) insoluble (S)	Table I Controlled Area Column 1 Air (pCi/ℓ)	Table I Controlled Area Column 2 Water (pCi/mℓ)	Table II Uncontrolled Area Column 1 Air (pCi/ℓ)	Table II Uncontrolled Area Column 2 Water (pCi/mℓ)
Cesium (55)	Cs 137	S	60	400	2	20
		I	10	1000	0.5	40
Cobalt (27)	Co 60	S	300	1000	10	50
		I	9	1000	0.3	30
Radon (86)	Rn 220	S	300		10	
	Rn 222	S	100		3	
Strontium (38)	Sr 90	S	1	10	0.03	0.3
		I	5	1000	0.2	40
Yttrium (39)	Y 90	S	100	600	4	20
		I	100	600	3	20

APPENDIX 7

EVALUATION OF RADIATION EXPOSURES

INTRODUCTION

A. Types of Radiation

Radiation is the emission or transmission of energy in the form of waves or particles. Examples are acoustic waves (i.e., sound), electromagnetic waves (such as radio, light, x- and gamma-rays), and particulate radiations (such as alpha particles, beta particles, neutrons, protons, and other elementary particles).

The class of radiation of importance to this report is known as ionizing radiation. Ionizing radiations are those, either electromagnetic or particulate, with sufficient energy to ionize matter, i.e., to remove or displace electrons from atoms and molecules. The most common types of ionizing radiation are x- and gamma-rays, alpha particles, beta particles, and neutrons.

X- and gamma-rays are electromagnetic waves of pure energy, having no charge and no mass or existence at rest. Gamma-rays and x-rays are identical except that x-rays originate in the atom and gamma-rays originate in the nucleus of an atom. X- and gamma-rays are highly penetrating and can pass through relatively thick materials before interacting. Upon interaction, some or all of the energy is transferred to electrons which, in turn, produce additional ionizations while coming to rest.

Alpha particles are positively charged particulates composed of two neutrons and two protons, identical to the nucleus of a helium atom. Due to its comparatively large mass and double charge, an alpha particle interacts readily with matter and penetrates only a very short distance before coming to rest, causing intense ionization along its path.

Beta particles are negatively charged free electrons moving at high speeds. Due to its comparatively small mass and single charge, a beta particle's penetration through matter is intermediate between that of the alpha particle and the gamma-ray, causing fewer ionizations per unit path length than an alpha particle.

B. Sources of Radiation

Ionizing radiations arise from terrestrial radioactive materials (both naturally occurring and man-made), extra-terrestrial (cosmic) sources, and radiation-producing machines. The sources of ionizing radiation important to this report are radioactive materials and cosmic sources.

Most atoms of the elements in our environment remain structurally stable. With time, an atom of potassium, for instance, may change its association with other atoms in chemical reactions and become part of other compounds, but it will always remain a potassium atom. Radioactive atoms, on the other hand, are

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not stable and will spontaneously emit radiation in order to achieve a more stable state. Because of this spontaneous transformation, the ratio of protons and neutrons in the nucleus of an atom is altered toward a more stable condition. Radiation may be emitted from the nucleus as alpha particles, beta particles, neutrons, or gamma-rays, depending uniquely upon each particular radionuclide. Radionuclides decay at characteristic rates dependent upon the degree of stability and characterized by a period of time called the half-life. In one half-life, the number of radioactive atoms and, therefore, the amount of radiation emitted, decrease by one half.

The exposure of man to terrestrial radiation is due to naturally occurring radionuclides and also to "man-made" or technologically enhanced radioactive materials. Several dozen radionuclides occur naturally, some having half-lives of at least the same order of magnitude as the estimated age of the earth. The majority of these naturally occurring radionuclides are isotopes of the heavy elements and belong to three distinct radioactive series headed by uranium-238, uranium-235, and thorium-232. Each of these decays to stable isotopes of lead (Pb) through a sequence of radionuclides of widely varying half-lives. Other naturally occurring radionuclides, which decay directly to a stable nuclide, are potassium-40 and rubidium-87. It should be noted that even though the isotopic abundance of potassium-40 is less than 0.012%, potassium is so widespread that potassium-40 contributes about one-third of the radiation dose received by man from natural background radiation. A major portion of the exposure (dose) of man from external terrestrial radiation is due to the radionuclides in the soil, primarily potassium-40 and the radioactive decay-chain products of thorium-232 and uranium-238. The naturally occurring radionuclides deposited internally in man through uptake by inhalation/ingestion of air, food, and drinking water containing the natural radioactive material also contribute significantly to his total dose. Many other radionuclides are referred to as "man made" in the sense that they can be produced in large quantities by such means as nuclear reactors, accelerators, or nuclear weapons tests.

The term "cosmic radiation" refers both to the primary energetic particles of extra-terrestrial origin that are incident on the earth's atmosphere and to the secondary particles that are generated by the interaction of these primary particles with the atmosphere, and reach ground level. Primary cosmic radiation consists of "galactic" particles externally incident on the solar system, and "solar" particles emitted by the sun. This radiation is composed primarily of energetic protons and alpha particles. The first generation of secondary particles (secondary cosmic radiation), produced by nuclear interactions of the primary particles with the atmosphere, consists predominantly of neutrons, protons, and pions. Pion decay, in turn, results in the production of electrons, photons, and muons. At the lower elevations, the highly penetrating muons and their associated decay and collision electrons are the dominant components of the cosmic-ray particle flux density. These particles, together with photons from the gamma-emitting, naturally occurring radionuclides in the local environment, form the external penetrating component of the background environmental radiation field which provides a significant portion of the whole-body radiation dose to man.

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In addition to the direct cosmic radiation, cosmic sources include cosmic-ray-produced radioactivity, i.e., cosmogenic radionuclides. The major production of cosmogenic radionuclides is through interaction of the cosmic rays with the atmospheric gases through a variety of spallation or neutron-capture reactions. The four cosmogenic radionuclides that contribute a measurable radiation dose to man are carbon-14, sodium-22, beryllium-7, and tritium (hydrogen-3), all produced in the atmosphere.

BACKGROUND RADIATION DOSES

Background radiation doses are comprised of an external component of radiation impinging on man from outside the body and an internal component due to radioactive materials taken into the body by inhalation or ingestion.

Radiation dose may be expressed in units of rads or rems, depending upon whether the reference is to the energy deposited or to the biological effect. A rad is the amount of radiation that deposits a certain amount of energy in each gram of material. It applies to all radiations and to all materials which absorb that radiation.

Since different types of radiation produce ionizations at different rates as they pass through tissue, differences in damage to tissues (and hence the biological effectiveness of different radiations) has been noticed. A rem is defined as the amount of energy absorbed (in rads) from a given type of radiation multiplied by the factor appropriate for the particular type of radiation in order to approximate the biological damage that it causes relative to a rad of x or gamma radiation. The concept behind the unit "rem" permits evaluation of potential effects from radiation exposure without regard to the type of radiation or its source. One rem received from cosmic radiation results in the same biological effects as one rem from medical x-rays or one rem from the radiations emitted by naturally occurring or man-made radioactive materials.

The external penetrating radiation dose to man derives from both terrestrial radioactivity and cosmic radiation. The terrestrial component is due primarily to the gamma dose from potassium-40 and the radioactive decay products of thorium-232 and uranium-238 in soil as well as from the beta-gamma dose from radon daughters in the atmosphere. Radon is a gaseous member of the uranium-238 chain. The population-weighted external dose to an individual's whole body from terrestrial sources in the United States has been estimated as 15 mrem per year for the Atlantic and Gulf Coastal Plain, 57 mrem per year for an indeterminate area along the Rocky Mountains, and 29 mrem per year for the majority of the rest of the United States. The overall population-weighted external dose for the U.S. population as a whole has been estimated to be 26 mrem per year.

The cosmic radiation dose, due to the charged particles and neutrons from secondary cosmic rays, is typically about 30% to 50% of the total from all external environmental radiation. The cosmic-ray dose to the population is estimated to be 26 mrem per year for those living at sea level, and increases with increasing altitude. Considering the altitude distribution of the U.S.

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population, the population-weighted external cosmic-ray dose is 28 mrem per year. The population-weighted total external dose from terrestrial plus cosmic sources is thus 54 mrem per year for the U.S. population as a whole.

The internal radiation doses derive from terrestrial and cosmogenic radionuclides deposited within the body through uptake by inhalation/ingestion of air, food, and drinking water. Once deposited in the body, many radioactive materials can be incorporated into tissues because the chemical properties of the radioisotopes are identical or similar to stable isotopes in the tissues. Potassium-40, for instance, is incorporated into tissues in the same manner as stable potassium atoms because the chemical properties are identical; radioactive radium and strontium can be incorporated into tissues in the same manner as calcium because their chemical properties are similar. Once deposited in tissue, these radionuclides emit radiation that results in the internal dose to individual organs and/or the whole body as long as it is in the body.

The internal dose to the lung is due primarily to the inhalation of polonium-218 and -214 (radon daughters), lead-212 and bismuth-212 (thoron daughters) and polonium-210 (one of the longer-lived radon decay products). The dose to the lung is about 100 mrem per year from inhaled natural radioactivity. The internal dose from subsequent incorporation of inhaled or ingested radioactivity is due to a beta-gamma dose from incorporation of potassium-40, rubidium-87, and cosmogenic nuclides, and an alpha dose from incorporation of primarily polonium-210, radium-226 and -228, and uranium-238 and -234. The dose to man from internally incorporated radionuclides is about 28 mrem per year to the gonads, about 25 mrem per year to the bone marrow, lung, and other soft tissues, and about 117 mrem per year to the bone (osteocytes). The bone dose arises primarily from the alpha-emitting members of the naturally occurring series, with polonium-210 being the largest contributor. The gonadal and soft tissue doses arise primarily from the beta and gamma emissions from potassium-40. The total internal dose from inhaled plus incorporated radioactivity is about 28 mrem per year to the gonads (or whole-body dose), about 125 mrem per year to the lung, about 25 mrem per year to the bone marrow, and about 117 mrem per year to the bone (osteocytes).

The total natural background radiation dose is the sum of the external and internal components. The population-weighted dose for the U.S. population as a whole is about 82 mrem per year to the gonads or whole body, about 179 mrem per year to the lung, about 79 mrem per year to the bone marrow, and about 171 mrem per year to the bone (osteocytes).

Besides the natural background radiation, background radiation doses include contributions from man-made or technologically enhanced sources of radiation. By far, the most significant are x-ray and radiopharmaceutical medical examinations. These contribute a population-averaged dose estimated to be 70 mrem per year for the U.S. population as a whole. Fallout from nuclear weapons testing through 1970 has contributed 50-year dose commitments estimated as 80 mrem external, and 30, 20, and 45 mrem internal to the gonads, lung, and bone marrow, respectively. Contributions from the use of fossil fuels (natural

APPENDIX 7
(cont'd.)

gas and coal) and nuclear reactors; mining, milling, and tailings piles; television sets, smoke detectors, and watch dials could be responsible for an additional 5 mrem per year, averaged over the U.S. population as a whole. In addition, the use of radiation or radioactivity for scientific, industrial, or medical purposes may cause workers in the industry and, to a lesser extent, members of the general public, to receive some radiation exposure above natural background.

EVALUATION OF RADIATION DOSE AND POTENTIAL HAZARD

Radiation, regardless of its sources, is considered to be a hazard because of its potential for producing adverse effects on human life. Very large amounts of radiation received over a brief period, i.e., hundreds of rem delivered within a few hours, can produce severe injury or death within days or weeks. Distributed over longer intervals, however, these same doses would not cause early illness or fatality. At dores and rates too low to produce these immediate symptoms, chronic or repeated exposure to radiation can bring about biological damage which does not appear until years or decades later. These low-level effects are stochastic in nature; their probability rather than their severity increases with dose. Primary among these latent or delayed effects are somatic effects, where insults such as cancers occur directly to the individual exposed, and genetic defects, where, through damage to the reproductive cells of the exposed individual, disability and disease ranging from subtle to severe are transmitted to an exposed person's offspring.

Clinical or observed evidence of a relationship between radiation and human cancers arise from several sources. The most important data come from the victims of Hiroshima and Nagasaki, patients exposed during medical therapy, radium dial painters, and uranium miners. Data exist only for relatively large doses; there have been no direct measurements of increased incidence of cancer for low-level radiation exposures. Evaluation of the available data has lead to estimates of the risk of radiation-induced cancer; estimated risks for the lower doses have been derived by linear extrapolation from the higher doses. All radiation exposures then, no matter how small, are assumed to be capable of increasing an individual's risk of contracting cancer.

Data on genetic defects resulting from radiation exposure of humans is not available to the extent necessary to allow an estimate of the risk of radiation-induced effects. Data from animals, along with general knowledge of genetics, have been used to derive an estimate of the risks of genetic effects.

Estimates of health effects from radiation doses are usually based on risk factors as provided in reports issued by the International Commission on Radiological Protection (ICRP), National Research Council Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR), or United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Multiplying the estimated dose by the appropriate risk factor provides an estimate of the risk or probability of induction of health effects to an individual or his descendants as a result of that exposure. The evaluation of these risk factors is presently subject to large uncertainties and, therefore, potential continual revision.

APPENDIX 7
(cont'd.)

The risk factors recommended by the ICRP for cancer mortality and hereditary ill health to the first and second generations are 10^{-4} per rem of whole-body dose and 4×10^{-5} per rem of gonadal dose, respectively. As an example, a whole-body dose of 1 rem would be estimated to add a risk of cancer mortality to the exposed invididual of 10^{-4}, i.e., 1 chance in 10,000. However, a precise numerical value cannot be assigned with any certainty to a particular individual's increase in risk attributable to radiation exposure. The reasons for this are numerous and include the following: (1) uncertainties over the influence of the individual's age, state of health, personal habits, family medical history, and previous or concurrent exposure to other cancer-causing agents, (2) the variability in the latent period (time between exposure and physical evidence of disease), and (3) the uncertainty in the risk factor itself.

To be meaningful, an attempt should be made to view such risk estimates in the appropriate context. One useful comparison is with risks encountered in normal life. Another comparison, potentially more useful, is with an estimation of the risks attributable to natural background radiation. Radiation from natural external and internal radioactivity results in the same types of interactions with body tissues as that from "man-made" radioactivity. Hence, the risks from a specified dose are the same regardless of the source. Rather than going through an intermediate step involving risk factors, doses can also be compared directly to natural background radiation doses.

Besides estimation of risks and comparisons to natural background, doses may be compared to standards and regulations. The appropriate standards, the Department of Energy's "Requirements for Radiation Protection," give limits for external and internal exposures for the whole body and specified organs which are expressed as the permissible dose or dose commitment annually in addition to natural background and medical exposures. There are, in general, two sets of limits, one applicable to occupationally exposed persons and the second applicable to individuals and population groups of the general public. The limits for individuals of the public are one-tenth of those permitted for occupationally exposed individuals. The set of limits important to this report are those applicable to individuals and population groups of the public. The limits for individuals of the public are 500 mrem per year to the whole body, gonads, or bone marrow and 1500 mrem per year to other organs. The limits for population groups of the public are 170 mrem to the whole body, gonads, or bone marrow and 500 mrem per year to other organs, averaged over the group. In either case, exposures are to be limited to the lowest levels reasonably achievable within the given limits.

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