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MA.02-8 MA.2

DOE/EV-0005/38

ANL-OHS/HP-83-108

**FORMERLY UTILIZED MED/AEC SITES
REMEDIAL ACTION PROGRAM
RADIOLOGICAL SURVEY
OF
THE FORMER WATERTOWN ARSENAL PROPERTY
GSA SITE
WATERTOWN, MASSACHUSETTS**



OCCUPATIONAL HEALTH AND SAFETY DIVISION
Health Physics Section
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

Operated by THE UNIVERSITY OF CHICAGO
for the U. S. DEPARTMENT OF ENERGY
under Contract W-31-109-Eng-38

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Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

NTIS price codes
Printed copy: A06
Microfiche copy: A01

Distribution Category:
Remedial Action and
Decommissioning Program
(UC-70A)

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ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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October 1983

Work performed under
Budget Activity DOE KN-03-60-40
and ANL 73706

PREFACE AND EXECUTIVE SUMMARY

This is one in a series of reports resulting from a program initiated in 1974 by the Atomic Energy Commission (AEC) to determine the condition of sites formerly utilized by the Manhattan Engineer District (MED) and the AEC for work involving the handling of radioactive materials. Since the early 1940s, the control of over 100 sites no longer required for nuclear programs has been returned to private industry or the public for unrestricted use. A search of MED and AEC records indicated that for some of these sites, documentation was insufficient to determine whether the decontamination work done at the time nuclear activities ceased is adequate by current guidelines. The Watertown Arsenal at Watertown, Massachusetts, is one such site. Therefore, radiological surveys of several areas within the Watertown Arsenal complex were conducted to ascertain current radiological conditions. The results of the radiological survey of Building Site 421 were contained in a final report (DOE/EV-0005/19) issued in February 1980. The results of a radiological survey of an area north of Arsenal Street, identified as the "GSA" property, are presented in this report. The results of radiological surveys of Building Site 34 and Building Site 41 are presented in a companion report (ANL-OHS/HP-83-106).

During the MED/AEC era, work involving radioactive materials was conducted at various sites within the arsenal complex. The GSA property covered in this report includes an area that had been used for packaging and storing of radioactive waste, burning of uranium scrap, and staging of radioactive-waste shipments during the MED/AEC period. This area contained most of the contamination found at the GSA site. Information provided by site personnel indicated that only depleted uranium was used at this GSA site. Results of radiological analyses of contaminated material taken from the site were consistent with this information. This area is still under Federal Government control.

During the period from July 13 through September 4, 1981, the Argonne National Laboratory (ANL), Occupational Health and Safety Division (OHS), Radiological Survey Group (RSG), conducted a comprehensive radiological survey at the GSA property at the request of the U.S. Department of Energy. Although the buildings at the GSA property were free from contamination, significant

levels of radioactive contamination were found on the site grounds, particularly in the area where radioactive uranium wastes had been burned. About 65 ft² of surface area exhibited elevated radiation levels. Soil borings indicated that the contamination extends to a depth of 6 ft at some locations and is in contact with groundwater. Hence there is potential for transport via suspended particles.

These levels of contamination are in excess of criteria as identified in ANSI 13.12 and NRC Guidelines.

This survey was performed under the auspices of the Health Physics Section of the Occupational Health and Safety Division of Argonne National Laboratory, Argonne, Illinois. The following personnel participated: R. A. Wynveen, W. H. Smith, R. L. Mundis (now at Los Alamos Scientific Laboratory), A. L. Justus, C. A. Hunckler, J. D. Thereon, R. Rodriguez, D. W. Reilly, and A. E. Lissy.

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FORMERLY UTILIZED MED/AEC SITES REMEDIAL ACTION PROGRAM
RADIOLOGICAL SURVEY OF THE FORMER WATERTOWN ARSENAL PROPERTY, GSA SITE
WATERTOWN, MASSACHUSETTS

INTRODUCTION

During the Manhattan Engineer District/Atomic Energy Commission (MED/AEC) era, work involving radioactive materials was conducted at various sites within the United States arsenal located at Watertown, Massachusetts (see Fig. 1). Portions of this property became contaminated, as a result of those activities. Subsequent decontamination was in accordance with standards and survey methods in use at that time. Since then, however, radiological criteria and guidelines for release of sites for unrestricted use have become more stringent, and documentation regarding the final radiological conditions at these sites on the arsenal properties is insufficient to establish conformity with present guidelines. Hence, the Argonne National Laboratory, Occupational Health and Safety Division, Radiological Survey Group, (ANL/OHS-RSG) was requested by the U.S. Department of Energy to perform radiological surveys of these properties to ascertain the extent of any remaining contamination.

The arsenal property included in the survey reported in this document consists of an area north of Arsenal Street (see Fig. 2) presently used by Federal agencies, principally the General Services Administration (GSA), but also including the U.S. Customs Service, the Bureau of Alcohol, Tobacco, and Firearms, and the Drug Enforcement Administration. Throughout this report, this area is referred to as the GSA property or site. This GSA property comprises about 12 acres of land and a number of buildings (see Fig. 3) being used for storage, equipment maintenance, and a pistol firing range. An outdoor fenced area is used for temporary storage of excess federal vehicles pending disposal at auction.

The portion of the GSA property located at the northern end of the fenced area (Fig. 3) had been used for packaging and storage of radioactive waste, burning of uranium scrap, and staging of radioactive waste shipments during the MED/AEC era. A small concrete pad remains at the former site of those operations. It was not possible to determine which of the buildings on the GSA

property had been used for work involving radioactive materials, so it was necessary to survey all the buildings, as well as the entire exterior property constituting the GSA site. Since the land on the northern and western sides of the fenced area was overgrown with weeds and scrub growth, a subcontractor was engaged to clear this growth so that a satisfactory radiological survey of the vacant land could be completed.

The ANL/OHS-RSG performed the survey of the GSA property during the period July 13 through September 4, 1981, and the results are reported herein. Surveys of other sites on the arsenal property are reported in two other documents. The former site of Building 421 (see Fig. 2) underwent a radiological assessment by the ANL/OHS-RSG in 1979. This area now comprises part of a park area with tennis courts. The results were published in a DOE report (DOE/EV-0005/19) issued in 1980. The property south of Arsenal Street, consisting of the former sites of Buildings 34 and 41 (Fig. 2), also underwent a radiological assessment, and results are being published in a companion report (ANL-OHS/HP-83-106).

SURVEY AND ANALYTICAL TECHNIQUES

General

The radiological assessment of the GSA property involved instrument surveys, to measure radiation levels of all accessible surface areas (both interior and exterior); smear surveys of surface areas where deemed appropriate; environmental soil sampling at representative locations; subsurface soil sampling and bore hole logging at select locations; air sampling at appropriate sites; sewer, water and sludge sampling where possible; and the determination of ambient radiation levels at the 3-ft level utilizing a pressurized ion chamber.

All the buildings on the GSA site (see Fig. 3) were surveyed in their entirety. For purposes of the assessment, the land area was gridded into reference squares 100-ft on a side as indicated in Figure 4. In addition, the land area was divided according to existing features into six convenient geographical sections (see Fig. 4). These divisions are as follows:

- Section I Secondly enclosed area at the northern end of the fenced property, formerly used for uranium burning.
- Section II Northern and central sections of the fenced property extending from beyond Building 237 to the edge of the paved area.
- Section III Paved area surrounding Buildings 234 and 235, extending to the fence line.
- Section IV Land area outside the west fence, extending from the fence line to the western property boundary.
- Section V Land area outside the north fence, extending from the fence line to the northern property boundary.
- Section VI Land area outside the east fence, extending from the fence line to the eastern property boundary.

The four buildings located on the GSA site and their current use are as follows:

- Building 234 Occupied jointly for storage by the U.S. Drug Enforcement Agency and the Federal Firearms, Tobacco, and Alcohol Tax Agency.
- Building 235 Utilized as a Federal pistol range (west one-third) and a GSA Warehouse (east two-thirds).
- Building 236 U.S. Customs storage, maintenance and office space.
- Building 237 GSA storage for flammable materials.

The exterior area that is enclosed by a secondary chain-link fence (Section I) includes a concrete pad that was the site of the uranium-burning and packaging operation during the MED/AEC operations.

Instrumentation

Four types of portable survey instruments were used to conduct the direct radiological surveys. Gas-flow proportional detectors with window areas of 51 cm² and 325 cm² (using Eberline PAC-4G-3 electronics) were used to monitor for alpha and/or beta-gamma radiation. NaI crystal detectors, 5 cm (2 in) diameter by 2-mm thick (Eberline PG-2 with Eberline PRM-5-3 electronics), were used to monitor for low-energy x-ray and gamma radiation. NaI crystal detectors, measuring 2.5 cm (1 in) diameter by 2.5 cm (1 in) thick (Eberline PRM-7 μ R meter) and calibrated with a ²²⁶Ra standard source, were used to measure the ambient external penetrating radiation field in units of μ R/h. An end-window Geiger-Mueller (GM) detector (Eberline HP-190 with a 7-mg/cm² window and Eberline 530 electronics), calibrated with a ²²⁶Ra standard source, was used to measure the contact exposure rate (mR/h) of contaminated areas. Integrated measurements of the ambient penetrating radiation field were taken with a Reuter-Stokes RSS-111 pressurized ionization chamber calibrated with an NBS traceable gamma-ray source. These instruments and associated calibration procedures are detailed in Appendices 1 and 2.

When possible, a contaminant was identified by performing gamma spectral analysis on the contaminated item, on a sample of material taken from the contaminated item, or on a sample of material taken from a contaminated area. These analyses were performed with a sodium iodide (NaI) or HyperPure Germanium (HPGe) detector coupled to a multichannel analyzer. This instrumentation is also described in Appendix 1.

Smear Surveys

Dry smears were taken at representative locations throughout each building with 4.25-cm-diameter filter papers (Whatman #1). A standard smear sample is obtained by applying moderate pressure with the tips of the first two fingers to the back of the filter paper and wiping the surface over an area of approximately 1000 cm². Smears were taken on original structures and components such as walls, floors, pipes, and vents. A smear of 100 cm² was taken from any area or object indicated by a portable survey instrument to have a higher than normal

radiation level. A smear of 100 cm² was also taken if the surface was extremely dusty.

To expedite the counting of the numerous smear samples collected, two counting techniques were employed with two types of counters. A large-area, thin-window, gas-flow proportional counter, sensitive to alpha and/or beta-gamma radiation, was used to make an initial count on groups of smears. For confirmatory counts on individual smears noted to be above the expected background level, a Nuclear Measurement Corporation Model PC-5 or 3A, internal gas-flow proportional counter (PC counter) with a thin aluminized Mylar window (referred to as Mylar spun top) was used.

Initial counts were made with the large-area counter on groups of ten smears at a time. Smears from any group indicating a reading above the instrument background were then counted individually in the PC counter. In addition, at least one smear from each group of ten was selected at random and counted in the PC counter. All smears of the areas or objects with elevated direct readings were counted individually in the PC counter. More detailed descriptions of the counters used and of the counting and calibration techniques used are presented in Appendix 1.

Air Samples

Air-particulate samples were collected using a commercial vacuum cleaner (ANL-modified) to pull air through filter media (Hollingsworth-Vose HV-70). A total volume of 26.7 m³ of air was sampled at a flow rate of 40 m³/h. A 10% portion (5 cm in diameter) was removed from the filter media after collection and counted for both alpha and beta-gamma activity in the PC counter. Concentrations of radon (²²²Rn), thoron (²²⁰Rn), and the presence of any long-lived airborne radionuclides were determined based on the result of several counts of each sample at specified intervals.

Air-particulate samples were also collected on Millipore membrane filter media for 40 minutes at a flow rate of approximately 1 m³/h. A portion of each filter sample was used for alpha spectral analysis to determine the actinon (²¹⁹Rn) concentration.

Details of air-sampling techniques and associated calculations are given in Appendix 3.

Soil Corings

Environmental soil samples (4-in-diameter by 12-in-deep corings) were taken from selected undisturbed locations (Fig. 8). Corings were taken at four additional sites away from the contaminated area (Fig. 7) to determine background levels of radionuclides in the soils of the area. Uranium fluorometric and gamma spectral analyses were conducted on all samples.

The samples were collected using a 4-in-diameter, 6-in-long right-circular-cylinder cutting tool, commonly used to cut holes on golf course greens. Each soil core was taken in four segments for analysis. Starting from the surface, three, 2-in segments were cut, bagged, and marked A, B, and C, respectively; the final segment of 6-in was marked D. The segmented coring technique was used to determine if any contaminant migration had occurred; to reduce the dilution of lower-level soil with the upper-level segments with respect to the surface deposition of the contaminants, or vice versa; and to reveal if any overburden or backfill had been added.

Soil Borings

Bore holes were drilled in areas exhibiting elevated radiation levels. Samples were taken from the hole in sequential 1-ft sections using a split-spoon sampler (1½-in inside diameter). The depth of the bore holes ranged to 6 ft. The bore holes were identified by a number (e.g., 1-S80), and each sample was identified according to depth in feet (e.g. 1-S80-2.0 equates to bore hole sample 1-S80 taken from 1.0 to 2.0 ft below the surface). Depths were reported to the nearest tenth of a foot.

Soil Analyses

Soil samples were prepared at ANL as detailed in Appendix 4 and shipped either to a commercial laboratory (LFE Environmental Analysis Laboratories) or to the Analytical Chemistry Section of the Chemical Engineering Division at Argonne National Laboratory for radiochemical and gamma-spectral analyses.

ANALYSIS OF SURVEY RESULTSGeneral

The measurements recorded with the gas-flow proportional-counter survey instruments were converted to surface contamination measurements according to the following general procedures (also see Appendix 2). For gross readings taken in the beta mode, background and any alpha contribution were subtracted to determine the net beta-gamma count rate. The net count rate then was converted to disintegrations per minute (dis/min) and normalized to a surface area of 100 cm². After subtraction of background, readings in the alpha mode also were converted to dis/min-100 cm². Two survey instruments were used. The first, identified as the floor monitor (FM), had a 325-cm² probe. The second, identified as a hand probe (HP), had a 51-cm² probe. 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.

Low-energy x-ray and gamma exposures were measured with the PRM-5-3 instrument. The results are reported in counts per minute (cts/min) and include the instrument background of 500 cts/min. The GM detector and μ R meter exposure-rate measurements include the instrument backgrounds of 0.03 mR/h and 7-10 μ R/h, respectively.

Smear samples were counted for both alpha and beta-gamma activity, and the net count rates are converted to dis/min-100 cm² after subtracting the appropriate background.

An average exposure reading was determined in the buildings where floor and wall surveys were performed. Additionally, an integrated background measurement of the ambient radiation field at select locations, both interior and exterior, was taken with a pressurized ion chamber (see Fig. 5).

Instrument measurements greater than the instrument backgrounds are presented in Table 1. The locations where these measurements were recorded are shown in Figure 6.

The instrument survey data and the smear results were reviewed with respect to both the ANSI Standard N13.12, "Control of Radioactive Surface Contamination of Facilities To Be Released for Uncontrolled Use," and the NRC's "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" (see Appendix 6).

Instrument and Smear Surveys

Buildings - No contamination was detected in the interior of any of the buildings on the GSA property. Approximately 75% of the walls and floors of the buildings were surveyed. Representative selections of the building overheads were subjected to both instrument and smear surveys. Smear samples were also taken at selected locations on the walls and floors. No radioactivity above background was found on any of the smears.

Each building was scanned at an elevation of about 1 m above the surface with a μ R gamma survey instrument (RPM-7). No instrument readings above the general background for the areas were observed. Integrated measurements of the ambient penetrating radiation field were also taken inside each building with a pressurized ionization chamber (Reuter-Stokes RSS-111). The results of these measurements are summarized in Table 2.

The floor plans for the four buildings (234, 235, 236, and 237), together with the location of the RSS-111 measurements, the air samples and the smear samples, are given in Figures 9-12.

Exterior Areas - The exterior areas of the site were subdivided into six survey sections identified as I through VI (see Fig. 4). The surveys for each section are discussed individually.

Section I - This area, which is enclosed by a secondary chain link fence, includes a concrete pad that reportedly had been used for passivating (i.e., burning) uranium metal scrap and packaging the resulting uranium oxide for disposal. Instrument surveys of the entire area were conducted using the four types of portable survey instruments described above. Contamination was found at 13 locations, identified as locations 34 through 46 (see Fig. 6). Each location was less than 100 cm² in area, so the total contamination did not exceed 1300 cm² in surface area. Alpha activity was

detected at locations 39, 40, and 45. Small metallic chips were found at locations 39 and 40. Preliminary gamma spectral analysis of the metal chips and surrounding soil indicated that the contaminant was other than natural uranium, either normal or depleted. Subsequent mass spectrometric analyses of several samples revealed that the contaminant was depleted uranium (see Table 6). The survey results for all the contaminated areas are presented in Table 1.

Ambient radiation levels were measured with a pressurized ionization chamber (Reuter-Stokes RSS-111) at two locations in this section (see Fig. 5). The results of these measurements are summarized in Table 2.

Section II - This area is composed of about the northern two-thirds of the fenced property (see Fig. 4). The ground surface is partly covered with cinders, and sparse vegetation grows over much of the area. Instrument surveys were conducted of the entire area with the four types of portable survey instruments described above. Contamination was found at 25 locations, primarily in the northern half of the section. Most of those locations were relatively small although one area covered about 5 m². The survey results for these contaminated areas are given in Table 1.

Results of ambient radiation levels measured with the pressurized ionization chamber at three locations in this section (see Fig. 5) are included in Table 2.

Section III - This section includes the paved areas surrounding the buildings located at the southern end of the fenced property (see Fig. 5). No significant contamination was found in the soil; however, a piece of wire with a reading about 3 mR/h at contact was found in the area. Preliminary gamma spectral analysis of the wire indicated that the contamination was due to ²²⁶Ra in equilibrium with its daughters. The general radiation levels for this section, as determined by the instrument survey, are given in Table 1.

Ambient radiation levels were measured with a pressurized ionization chamber at three locations in this section also (see Fig. 5). The results are given in Table 2.

Section IV - This section extends from the west fence line to the west property line and from the south property line (along Arsenal Street) to the extension of the north fence line (see Fig. 4). This area includes the remnants of a partially paved driveway (along the fence line), a swampy area adjacent to the small creek, and several piles of "fill" material scattered in various locations. The entire area was overgrown with brush, which was cleared by a local landscape contractor prior to the survey. All brush and debris were surveyed for contamination prior to disposal, and all equipment used in the cleaning operation was surveyed after use. No radioactive contamination was detected. Although the radiological survey of the area revealed noticeable variability in the instrument readings, depending on the type of surface, no explicit contamination was detected. The general radiation levels for this section, as determined by the instrument survey, are given in Table 1.

A single measurement of the ambient radiation level was made with the pressurized ion chamber in this section (see Fig. 5). The results of this measurement are included in Table 2.

A rock outcropping near the property boundary in this section (i.e., grid F-1) indicated anomalously high radioactivity. A small sample (identified as 1-R96 in Table 4) of this rock was taken for analysis. The radioactivity was due to natural uranium and natural thorium.

Section V - This area includes all the property outside the north fence line, extending to the property boundary lines on the east, west, and north (see Fig. 4). This section also was heavily overgrown with brush and was cleared in the same fashion as described above for Section IV. All larger trees were left standing. No radioactive contamination was found on any of the brush, debris, or equipment. While the majority of this area was uncontaminated, a few localized areas (totaling less than 1800 cm²) had somewhat elevated radiation levels. These direct survey results are summarized in Table 1. The possibility exists that the elevated radiation levels in these areas result from natural radioactivity in fill material used, rather than contamination from former MED/AEC activities.

Ambient radiation levels were measured with a pressurized ionization chamber at three locations in this section (see Fig. 5), results of these measurements are included in Table 2.

Section VI - This section includes all the property outside the east fence line extending to the south boundary along Arsenal Street, the east boundary along Greenough Boulevard, and the extension of the north fence line (see Fig. 4). The presence of a marshy area in the northern third of this section precluded a complete (100%) surface survey of the section. No contamination was detected in this section and no ambient radiation levels were measured.

Air Samples

"Grab" air samples were collected at 14 locations within the four structures located on the GSA site. The results are presented in Table 3 and locations where each air sample was taken are shown in Figures 9 through 12. The detailed calculations used in evaluating the air samples are given in Appendix 3. The radon (^{222}Rn) daughter Working Levels (WL) in the buildings ranged from 0.0004 WL to 0.0102 WL. These values are well below the limit of 0.02 WL for average annual concentration as specified in the EPA Standard (40 CFR 192). Thoron (^{220}Rn) levels, as estimated by counting the air samples after the radon (^{222}Rn) had decayed (see Appendix 3), were very low.

A separate air sample using Millipore membrane filter paper was taken in each building (samples 18 and 20 through 22) to determine if actinon (^{219}Rn) was present in the air (see Appendix 3); none was detected in any of these air samples.

Soil Corings

Environmental soil samples (soil corings) were taken at 23 locations throughout the site. The locations of these samples were chosen so as to encompass the areas identified as potentially contaminated by the surface surveys. The location of each soil coring is shown in Figure 8. The soil corings have been identified as 1-S47 through 1-S62, 1-S97 through 1-S101, 1-S103 and 1-S104.

Soil scrapings identified as 1-S76, 1-S102, 1-S105, and 1-S106, were also taken from contaminated locations (see Fig. 8). All soil corings were sectioned and analyzed for uranium (uranium fluorometric) as well as radium and thorium decay chains (gamma spectral analysis) as described in Appendix 4. The results of these analyses are presented in Table 4.

Two background soil corings were taken at each of two offsite locations (see Fig. 7) to determine background levels of radionuclides in soil of that region of Massachusetts. These corings, identified as 1-SB107 through 1-SB110, were analyzed as above and the results are also reported in Table 4. These background soil samples indicated natural uranium concentrations ranging from 0.9 to 4.6 pCi/g, with one exception. Sample 1-SB110-B had 11.8 pCi/g of natural uranium. The high value is probably a result of soil fertilization. Since elevated concentrations of uranium are present where phosphate-containing fertilizers are used, any fertilizing of the soil can result in an increased uranium concentration.

As shown in Table 4, the results of the analyses of soil corings from the GSA site indicated several areas with significantly high levels of contamination. Samples 1-S47 through 1-S50, 1-S54, 1-S62, 1-S99 and 1-S102 through 1-S106, all showed elevated levels of uranium. (Samples 1-S102, 1-S105 and 1-S106 were surface scrapings.) None of the samples showed elevated levels of radium (or thorium) as determined by gamma spectral analysis of the radium and thorium decay chains (see Table 4). Based on these findings, the contamination consists of other than natural uranium (i.e., uranium that at some time in the recent past was chemically separated from its daughter elements). The contamination was as high as 2.6×10^4 pCi/g (1-S104), with three samples having levels in excess of 10^4 pCi/g (1-S104 through 1-S106). Of the remaining nine contaminated samples, five (1-S47 through 1-S50 and 1-S99) had levels considerably in excess of 100 pCi/g. The locations from which these contaminated corings were collected is shown in Figure 8.

Soil Borings

Subsurface investigations using soil-boring techniques also were conducted within the contaminated areas at the site in order to determine the vertical profile of the contamination. A total of 19 bore holes were drilled to a depth of 6 ft. It was not possible to drill deeper than 6 ft because groundwater was

encountered at that level. At 6 of the 19 bore holes drilled there was still significant contamination at the 6-ft level. The bore holes, identified as 1-S77 through 1-S95, are located in Figure 13. Split-spoon samples were taken from each bore hole at continuous 1-ft increments.

Bore-hole logging was accomplished using a 2-in x 2-in NaI(Tl) detector in conjunction with a ND-100 multichannel analyzer. Readings were taken at grade level and at 2-ft increments thereafter. The results of the bore hole logging did not reveal any unexpected anomalies.

Each soil sample was prepared as outlined in Appendix 4 and then analyzed for uranium (uranium fluorometric) as well as radium and thorium decay chains (gamma spectrometric). The results of these analyses are given in Table 4. The absence of radium (^{226}Ra) decay chain in most of the contaminated samples indicated that the bulk of the contamination was due to other than natural uranium (i.e., uranium that had been chemically separated from its daughters).

Mass spectrometric analyses were performed on several of the more highly contaminated soil samples. These measurements were made to determine whether the uranium contamination resulted from depleted uranium (i.e., uranium depleted in the ^{235}U isotope as a result of isotope separation) as had been reported. The results of these measurements are given in Table 6. All the samples except 1-R96 (see Fig. 8) were about threefold depleted in the ^{235}U isotope, confirming that the contamination is the result of depleted uranium. Sample 1-R96 was from a rock outcropping and is natural uranium and natural thorium indigenous to the site. This sample also had the equilibrium concentration of ^{226}Ra (see Table 4) as would be expected for natural uranium.

Analyses of the soil samples taken from the bore holes revealed significant levels of contamination (see Table 4). Two of the bore holes (1-S92 and 1-S95) contained elevated amounts of ^{226}Ra (decay chain), indicating the presence of ^{226}Ra with daughters or "tailings" material (i.e. material remaining after the uranium has been chemically separated). The highest concentration was found in the first 2 ft of hole 1-S92, where about 14 pCi/g was found. Although the concentration of the contamination in hole 1-S95 was less (about 5 pCi/g), contamination was found to descend to the 6-ft level.

Three of the holes (1-S87, 1-S91, and 1-S94) contained contamination that showed equilibrium concentrations of the ^{226}Ra decay chain, indicating the presence of natural uranium (i.e. uranium in equilibrium with all its natural

daughters). This contamination was relatively light (up to 12 pCi/g of uranium) and was restricted to the first couple of feet of soil.

Thirteen of the remaining holes (1-S77 through 1-S86 and 1-S88 through 1-S90) had other than natural uranium contamination. This contamination reached as high as 588 pCi/g (1-S85-2) and was found as deep as the 6-ft level in five of the holes (1-S80, 1-S81, and 1-S83 through 1-S85). The most severely contaminated holes (1-S81 through 1-S86) were located in the relatively small area identified in Section I (see Figs. 4 and 13).

Water and Sludge Samples

All sewer lines and natural surface drainage paths serving the GSA property were sampled to the extent possible. Upstream and downstream samples of both water and bottom sludge were obtained from the small stream that follows the southern boundary of the property. Water and bottom sludge samples were also taken from a swampy area north of the property. This swamp seems to be in the drainage path for the groundwater flowing from the site toward the Charles River. The locations for all these water and sludge samples are shown in Figure 14.

Samples of groundwater were also taken from bore holes 86, 90, 91, 92, 93, and 95. These water samples were labeled 1-W86, 1-W90, 1-W91, 1-W92, 1-W93, and 1-W95, respectively.

All water samples were separated into suspended solids and dissolved solids fractions; then each fraction was analyzed for uranium (uranium fluorometric) and ^{226}Ra and ^{232}Th decay chains (gamma spectrometric). The results are reported in Table 5. No anomalous levels of contamination were found in any of the dissolved solids fractions. However, some elevated levels were found in the suspended solid fractions from some of the samples. A slightly elevated level of uranium was found in sample 1-W69, which came from outside the fence (see Fig. 14). Somewhat higher levels of uranium contamination were found in sample 1-W91 (taken from bore hole 91) and a rather high level of uranium (55 pCi/g) was found in sample 1-W86 taken from bore hole 86 (see Table 5 and Fig. 13). Since the ^{226}Ra decay chain levels were normal in all these samples, this contamination is judged to be other than natural (probably depleted) uranium. These levels of contamination in the suspended solids are to be expected since

the soil in these bore holes is contaminated. Nevertheless, the results indicate that the uranium contamination is being transported via suspended particles by the groundwater.

ESTIMATED EXTENT OF CONTAMINATION

No contamination was detected in the interior of any of the buildings on the site. Surface contamination was detected in three of the six exterior areas delineated in Fig. 4. Most of the contamination was found in Sections I and II, with significantly less in Section V and none in Sections III, IV, and VI. The total contaminated surface area was less than 1300 cm² in Section I, about 5.5 m² in Section II, and less than 1800 cm² in Section V, for a total of about 6 m².

Soil corings and scrapings were taken from 27 locations throughout the site (see Fig. 8). Twelve of the corings had elevated levels of radioactivity. The contamination was determined to be depleted uranium with levels of radioactivity ranging up to 2.6×10^4 pCi/g (1-S104).

Bore holes were drilled at 19 locations (see Fig. 13) to a depth of 6 ft (the level of the water table). Contamination was found as deep as 6 ft in six of the bore holes. The highest level of contamination found in the bore holes was about 600 pCi/g of uranium at the 2-ft level in hole 1-S85. The contaminant was predominantly depleted uranium.

For this evaluation, the following assumptions have been made:

1. Six square meters of area is contaminated.
2. The depth of contamination will be averaged (Option #1) and maximized (Option #2). The average depth of contamination is 3 ft and the maximum depth is 6 ft.
3. The concentration of the contamination will also be averaged (Option #1) and maximized (Option #2). The average concentration of the depleted uranium contamination is taken as 240 pCi/g and the maximum is taken as 2.6×10^4 pCi/g based on uranium analyses of the soil samples (see Table 4).

Based on these assumptions, the volume, mass, and radioactivity of the material involved have been calculated and are reported in Table 7. As indicated in the table, the extent of contamination could range from 6 m³ of material with a mass of 9000 kg and radioactivity of 240 pCi/g (Option #1 assumptions) to 12 m³ of material with a mass of 18,000 kg and radioactivity of 2.6 x 10⁴ pCi/g (Option #2 assumptions). Details of these calculations is given in Appendix 7.

DOSE AND POTENTIAL HAZARD EVALUATION

External Exposure

To assess the radiological hazard from external exposure to the radiation sources, a "conservative" (or worst-case) situation was assumed. Since commercial, rather than residential occupancy is involved, it was assumed that an individual would be exposed 40 hours per week, 50 weeks per year, to the maximum radiation level. The maximum radiation level observed was 7 mR/h (contact reading) at location N-2 in Section II. The annual radiation dose from this source would be:

$$7 \text{ mR/h} \times 40 \text{ h/w} \times 50 \text{ w/y} \times 1 \text{ rem/R} = 14 \text{ rem/y}$$

This dose calculation is based on the assumption that a person is in direct contact with this source of exposure continuously for 40 hours per week, an extremely unlikely situation. The resulting dose is well above the DOE 5480.1⁽¹⁾ limit of 500 millirem per year for a person non-occupationally exposed (see Appendices 6 and 8).

A more realistic value for the dose from external exposure would be based on the maximum radiation level determined at the 3-ft level, which was about 0.04 mR/h in Section I. The annual radiation dose based on this value would be:

$$0.04 \text{ mR/h} \times 40 \text{ h/w} \times 50 \text{ w/y} \times 1 \text{ rem/R} = 80 \text{ millirem}$$

This dose is well below the 500 millirem limit referred to above.

Internal Exposure

To assess the potential for radiological hazard based on potential internal exposure, it was necessary to assume some "conservative" but nevertheless plausible scenarios whereby the radioactive contamination was assimilated internally. To this end, two hypothetical cases were considered. The first case was based on the assumption that a child would eat 100 g of contaminated soil per year. For the second case, a person was assumed to rototill the contaminated soil (dry) to a 1-ft depth for a working day (8 hours) once a year. For this latter case, a resuspension factor of 10^{-6} and a breathing rate of $9.6 \text{ m}^3/\text{working day}$ were used.⁽²⁾ In both cases it was assumed that the average concentration of contaminants in the soil was equal to the maximum measured value (a conservative assumption). All calculations are based on methods outlined in ORNL/NUREG/TM-190, Vol. 3.⁽³⁾ These calculations approximate the ICRP-30 guidelines for hazard analysis.⁽⁴⁾ Details of the calculations are given in Appendix 8.

The maximum concentration of soil contaminant found was $2.6 \times 10^4 \text{ pCi/g}$ depleted uranium (i.e., uranium exclusive of its natural daughters and depleted in the isotope ^{235}U). Based on this level of contamination, the following hazard levels (50-year dose commitment for one year of intake) were calculated:

1. Child eating 100 g per year

(Units: $\text{mrem/pCi} \times \text{pCi/g} \times \text{g} = \text{mrem}$)

Depleted uranium:	Bone	$(2.8 \times 10^{-4})(2.6 \times 10^4)(100)$	= 0.7 rem
	Total body	$(2.1 \times 10^{-5})(2.6 \times 10^4)(100)$	= 55 mrem

2. Adult inhalation of aerosol (per year intake)

(Units: $\text{mrem/pCi} \times \text{pCi/m}^3 \times \text{m}^3 = \text{mrem}$)

Depleted uranium:	Bone	$(7.1 \times 10^{-3})(1.2 \times 10^4)(9.6)$	= 0.8 rems
	Total body	$(1.5 \times 10^{-2})(1.2 \times 10^4)(9.6)$	= 1.7 rems
	Lung	$(0.48)(1.2 \times 10^4)(9.6)$	= 55 rems

These levels would certainly pose a significant radiological hazard. However, this worst-case scenario is not representative of the general condition at the site. Hence there is not cause for grave concern. Nevertheless, care should be taken when any excavation-type work is undertaken or if children are allowed to play in the area (see Appendix 8).

CONCLUSIONS

The complete radiological survey of all the buildings located on the GSA site revealed no contamination. However, significant contamination was found on the site grounds. A total of approximately 6 m² of surface area exhibited elevated radiation levels. The area of contamination is centered around Section I (the burn area) and spreads out from this point (see Fig. 13). The contamination includes the subsurface, descending to the water table at the 6-ft level in some places. The contamination is primarily reprocessed depleted uranium.

The ambient radiation level in the vicinity of the contamination as measured with a pressurized ionization chamber is significantly above background levels (up to 18 μ R/h, Table 2). As expected, no significant levels of radon contamination were found in any of the air samples (Table 3).

These levels of contamination exceed the criteria as defined in ANSI 13.12 and the NRC Guidelines (see Appendix 6). The ambient radiation levels (up to 18 μ R/h) are not in excess of criteria as defined in Appendix 6.

REFERENCES

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2. U.S. Department of Health, Education and Welfare. 1970. "Radiological Health Handbook." Bureau of Radiological Health, Rockville, MD.
3. U.S. Nuclear Regulatory Commission. 1981. "Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities." Vol. III. ORNL/NUREG/TM-190. Vol. 3. Prepared by Oak Ridge National Laboratory.
4. International Commission on Radiological Protection. 1979. "Limits for Intakes of Radionuclides." ICRP Publication 30 (Pergamon Press, New York)

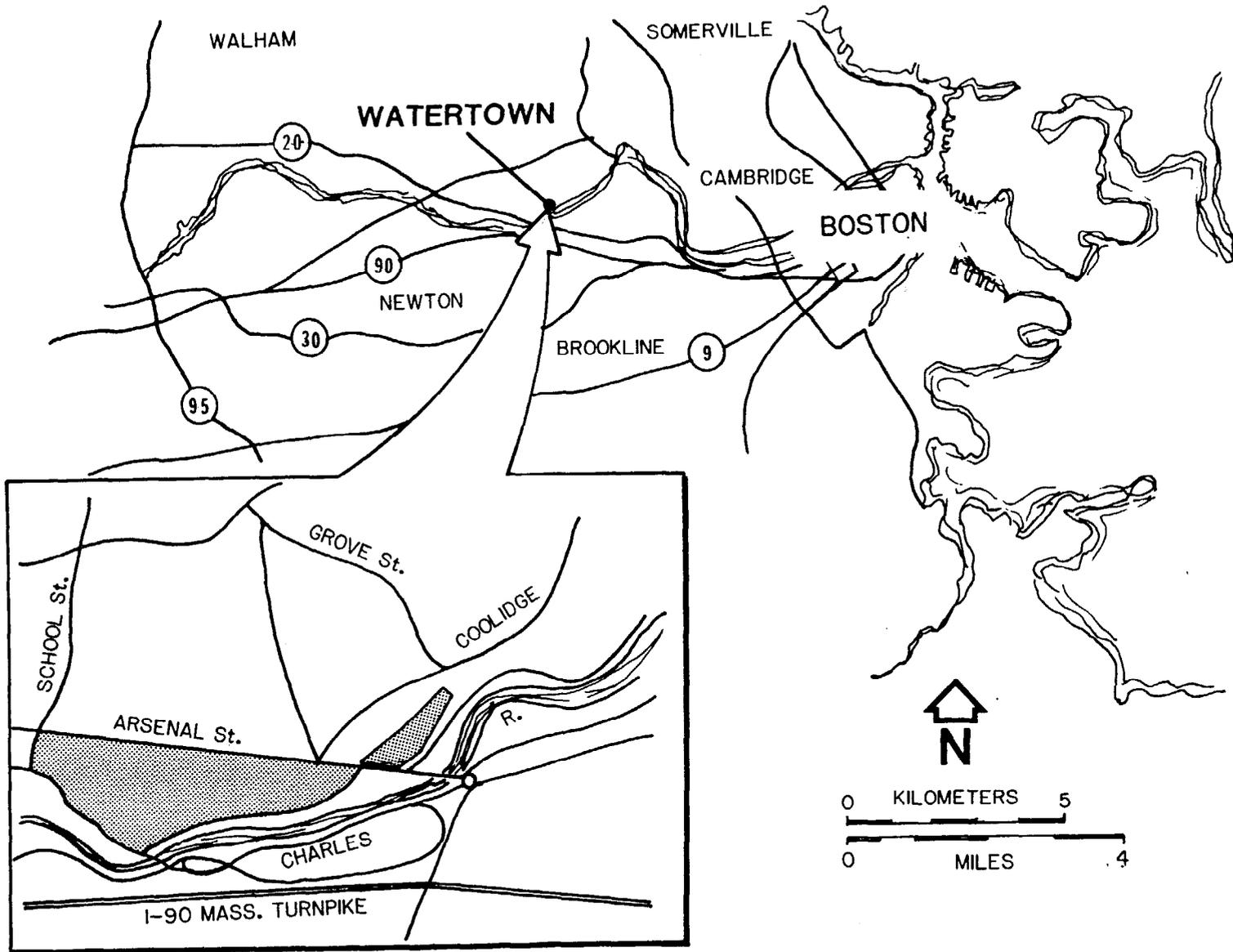


Fig. 1. Watertown Arsenal Site Area.
ANL-HP Dwg. No. 82-10

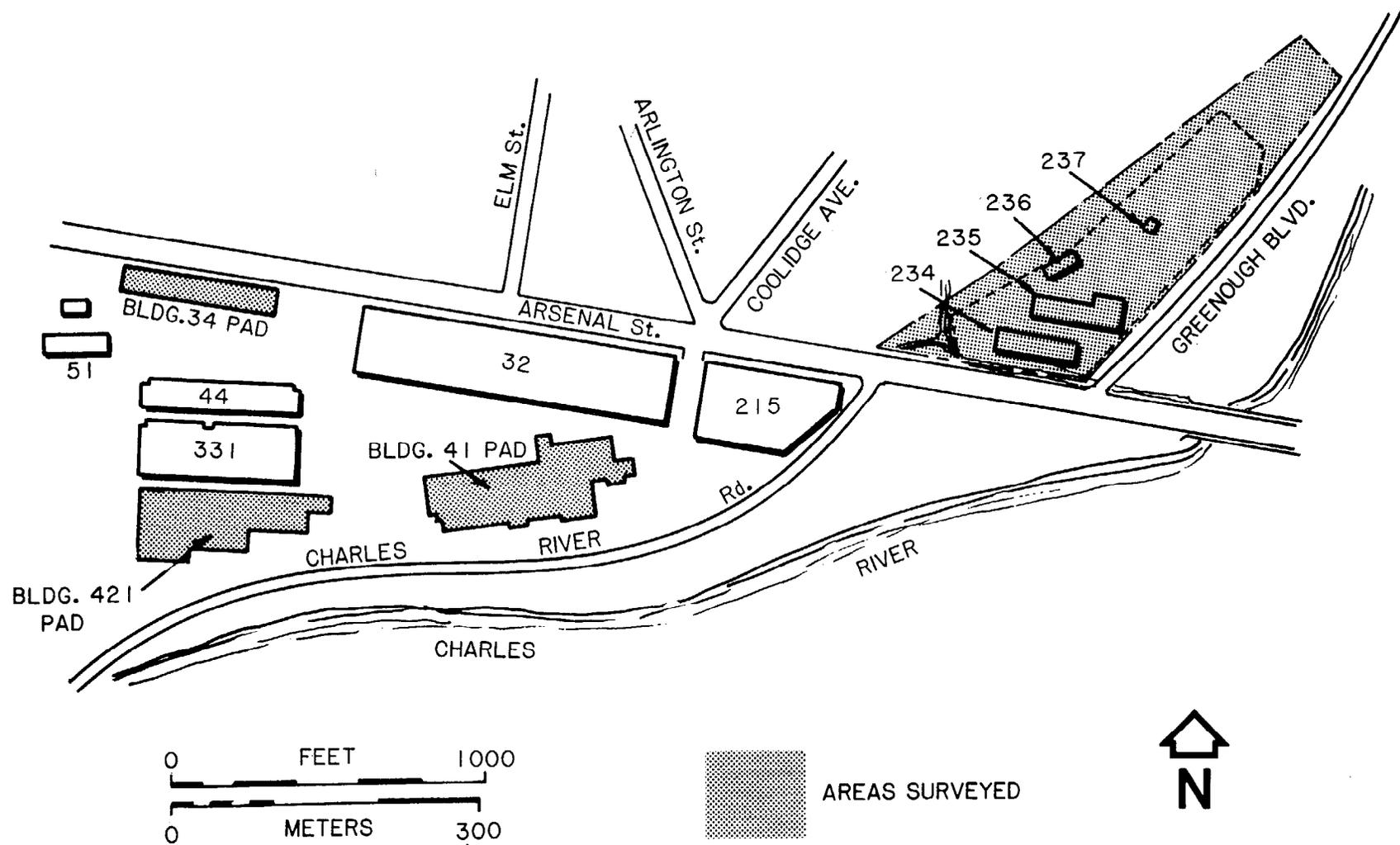
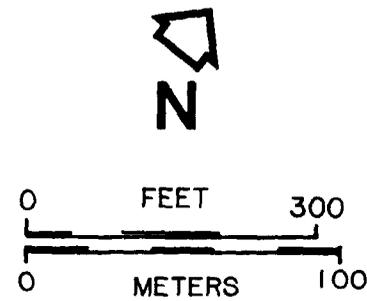
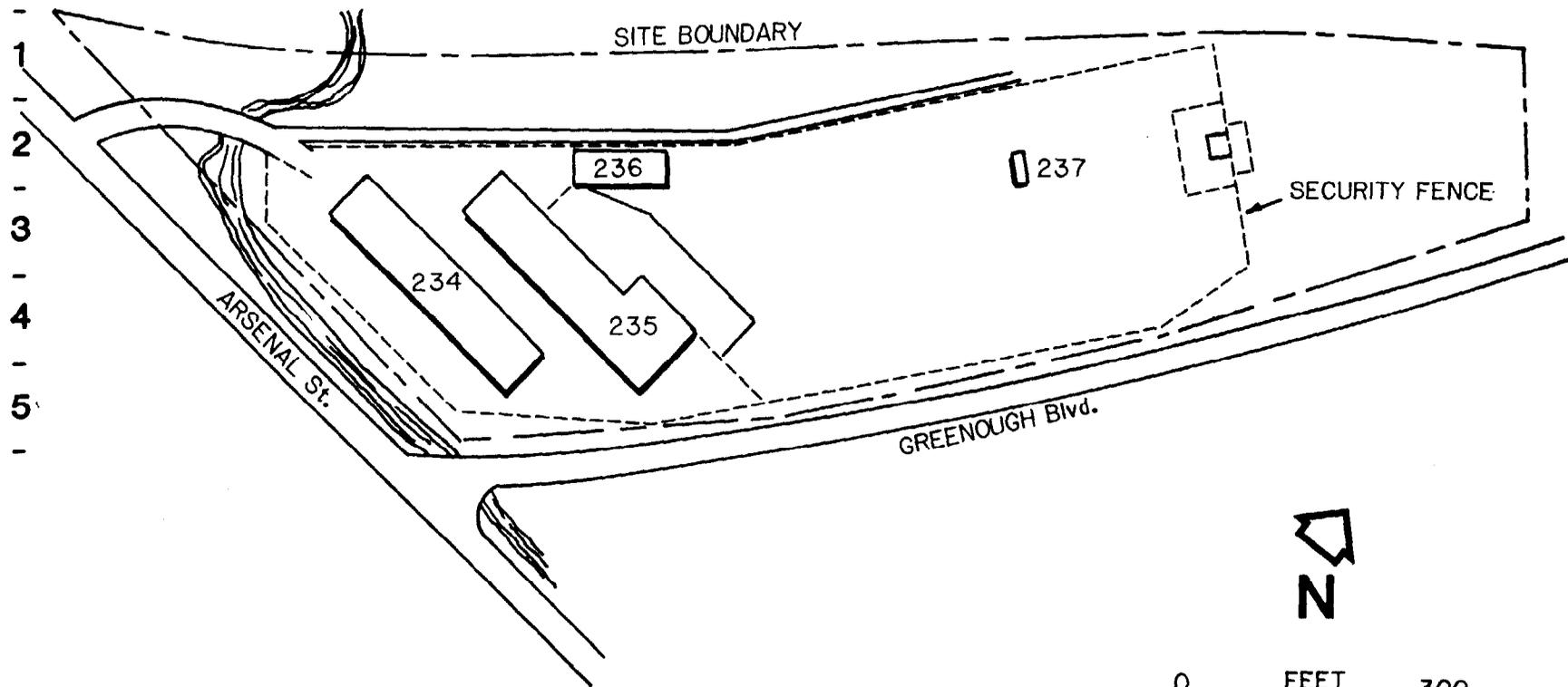
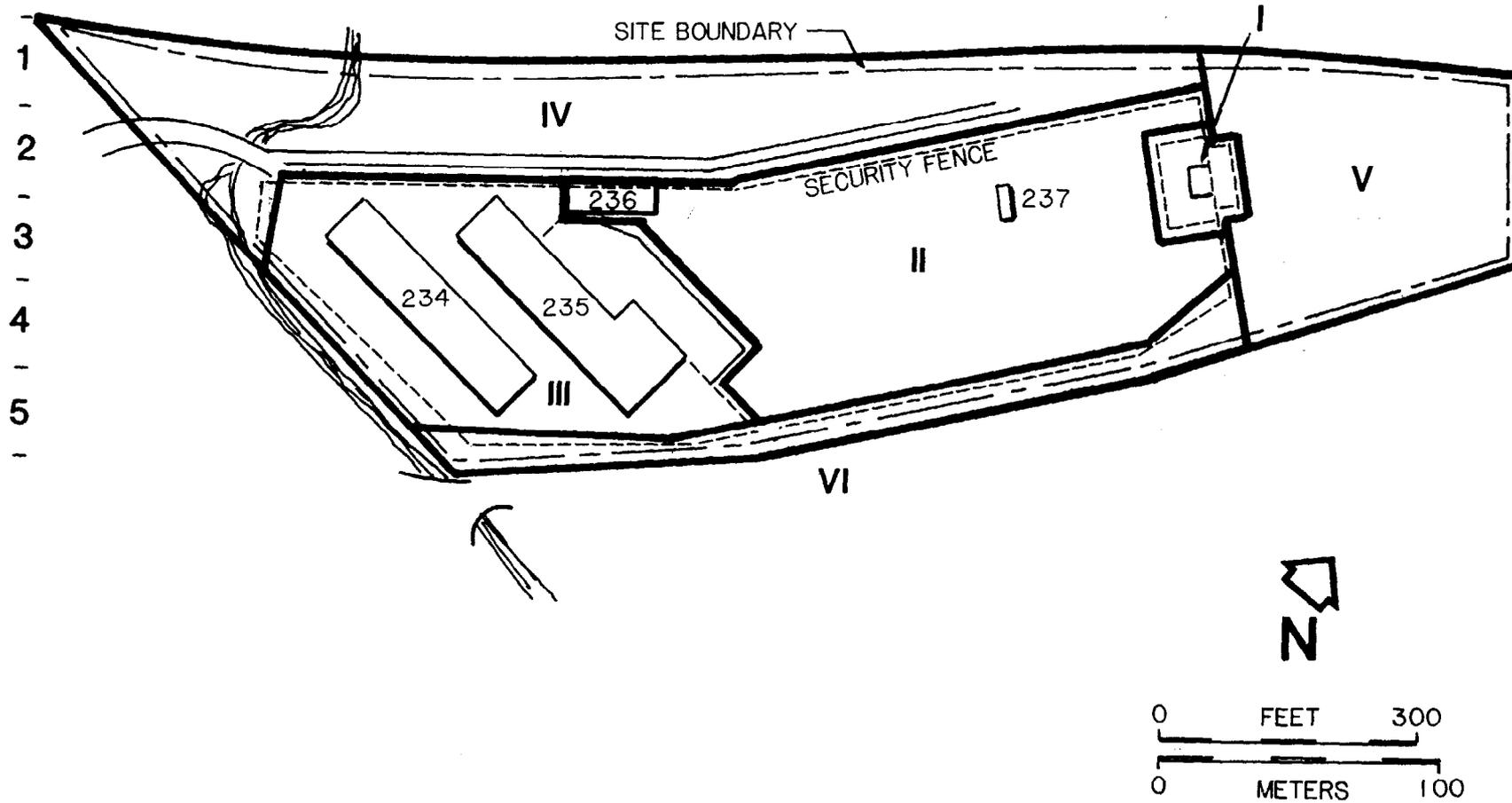


Fig. 2. Watertown Arsenal - Survey Site.
ANL-HP Dwg. No. 82-6.



A | B | C | D | E | F | G | H | I | J | K | L | M | N | O | P | Q |

Fig. 3. North Area GSA Property.
ANL-HP Dwg. No. 82-7



23

| A | B | C | D | E | F | G | H | I | J | K | L | M | N | O | P | Q |

Fig. 4. Survey Subdivisions - GSA Property.
ANL-HP Dwg. No. 83-66

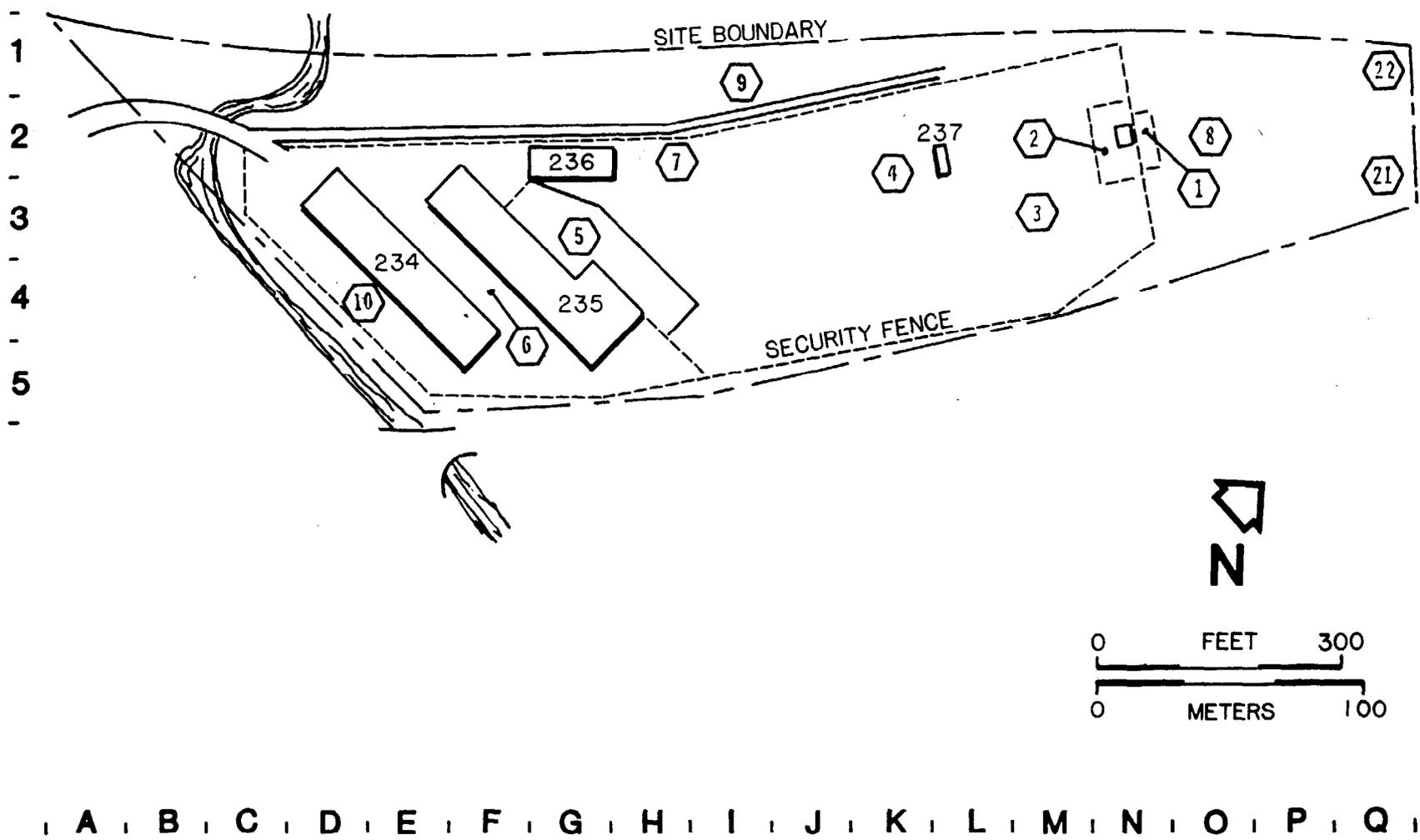
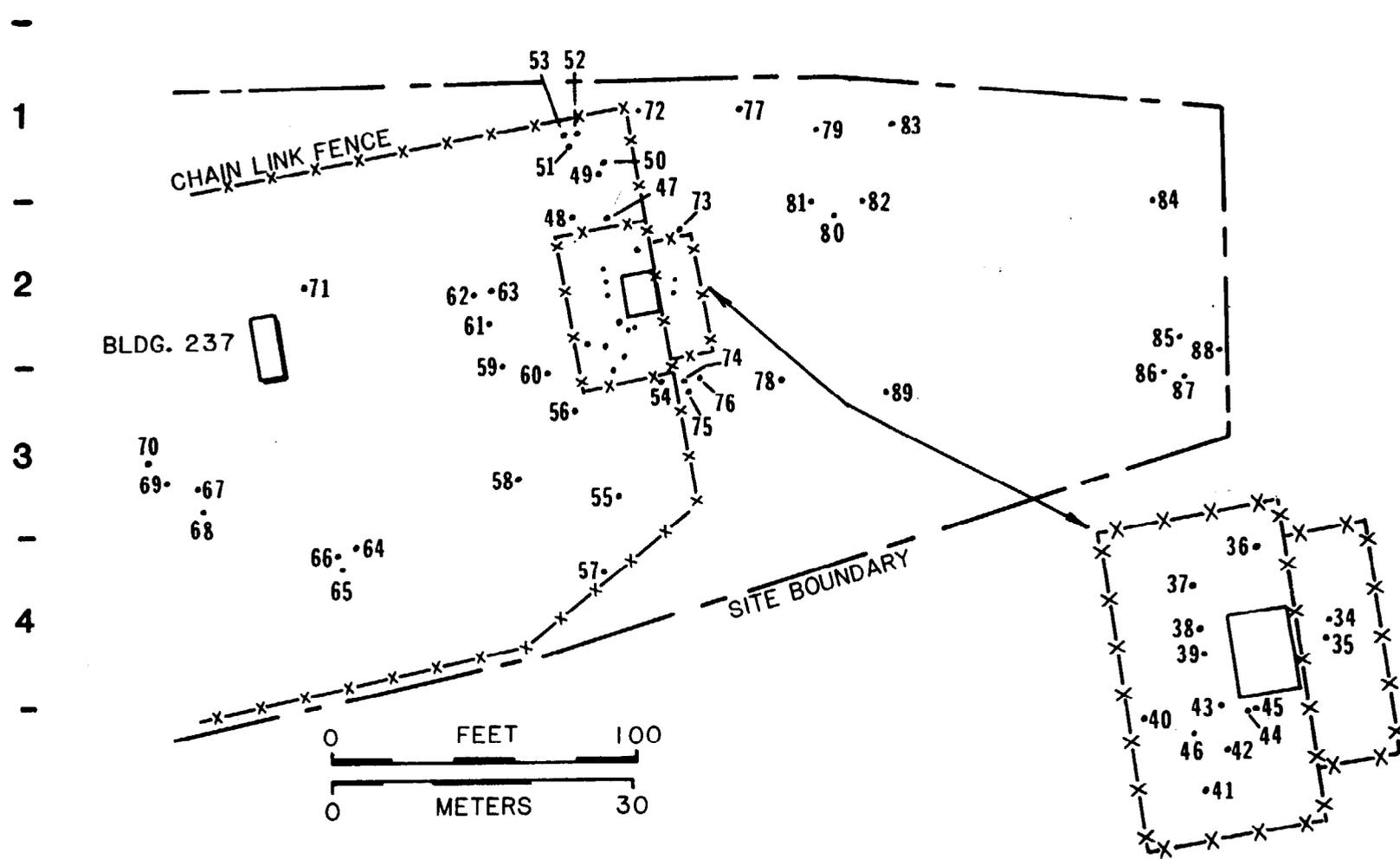


Fig. 5. Gamma Background Measurement Locations.
ANL-HP Dwg. No. 82-8



I K I L I M I N I O I P I Q I

Fig. 6. Location of Contamination.
ANL-HP Dwg. No. 83-69

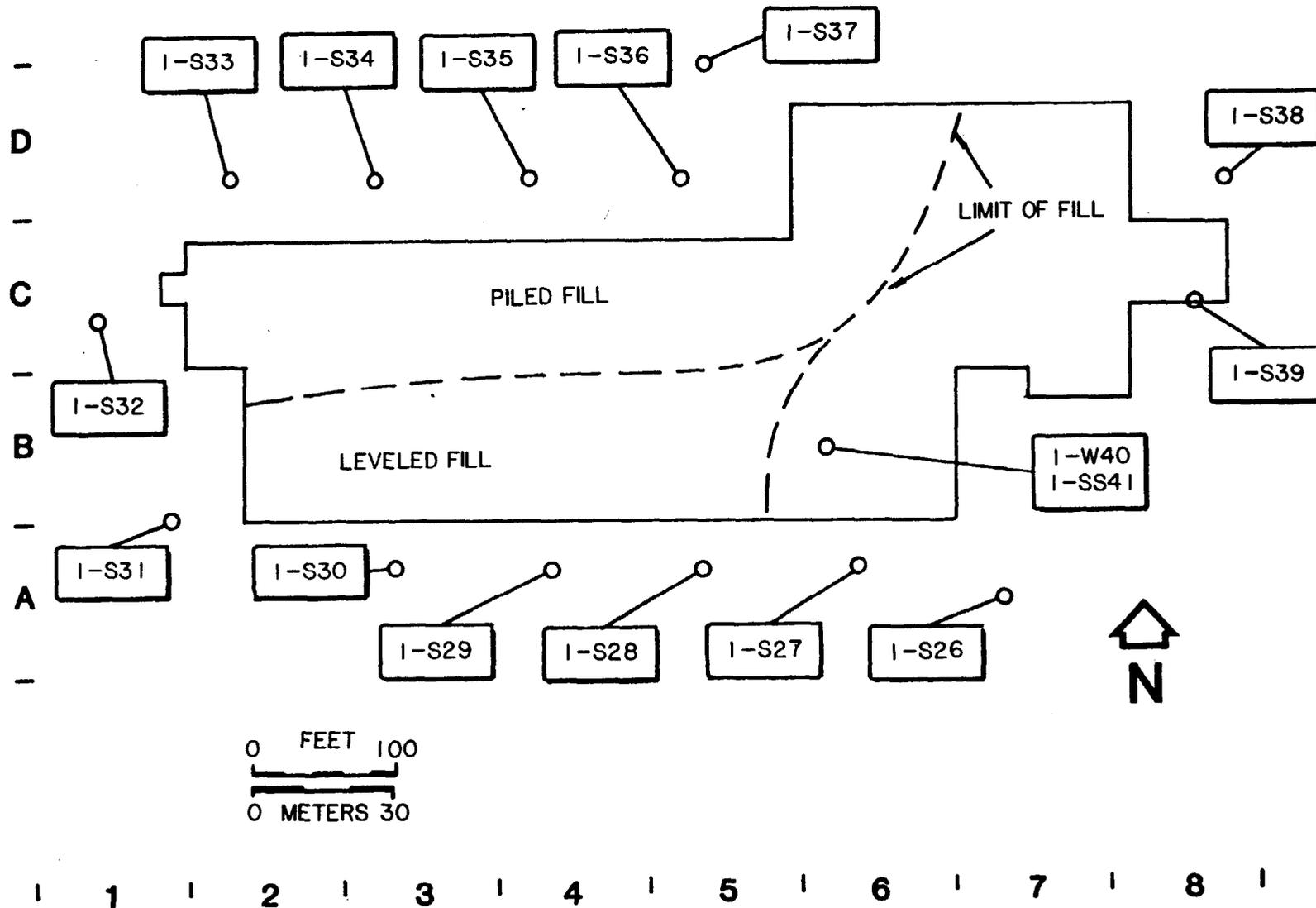


Fig. 7. Background Soil Coring Locations.
ANL-HP Dwg. No. 78-7

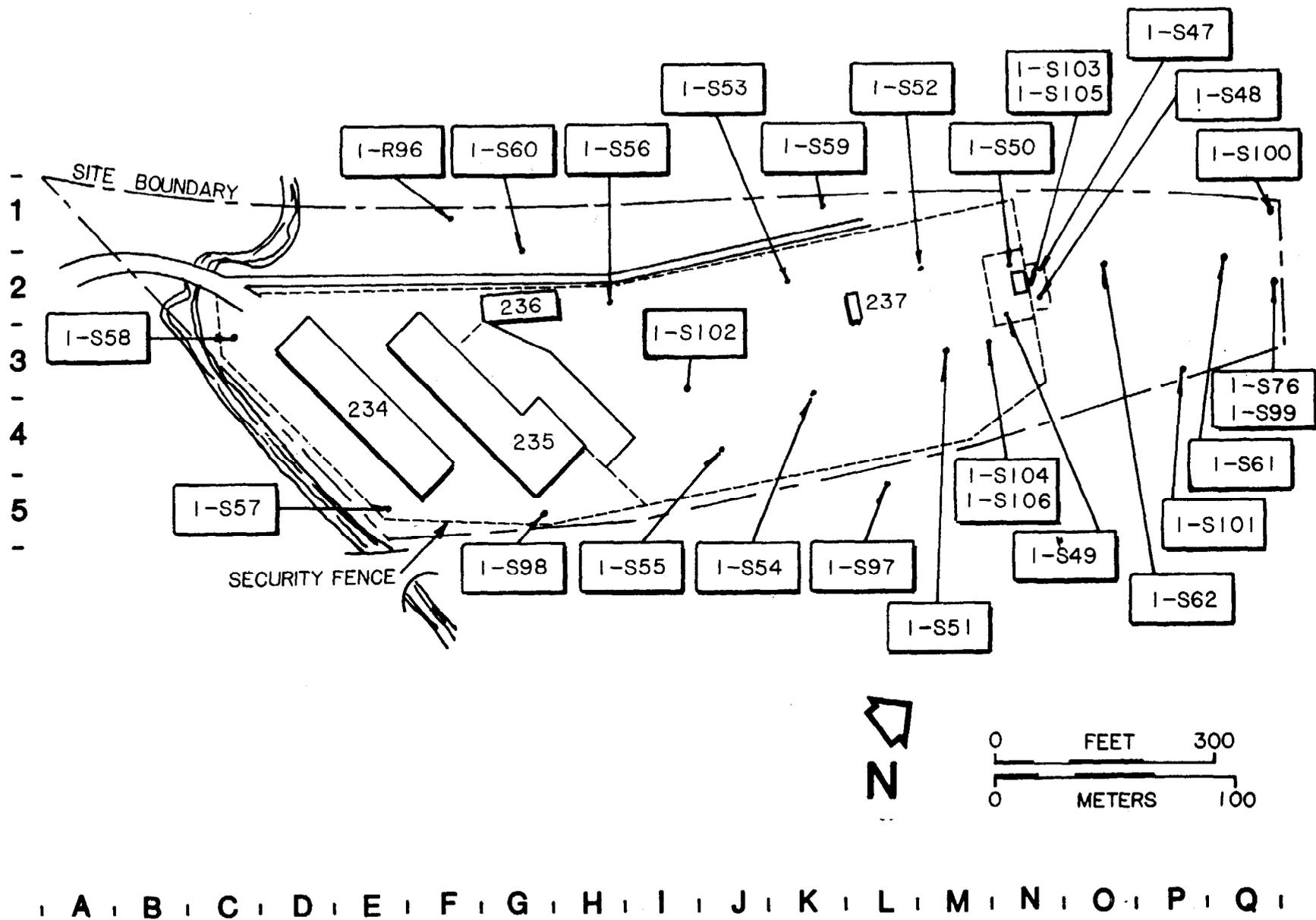
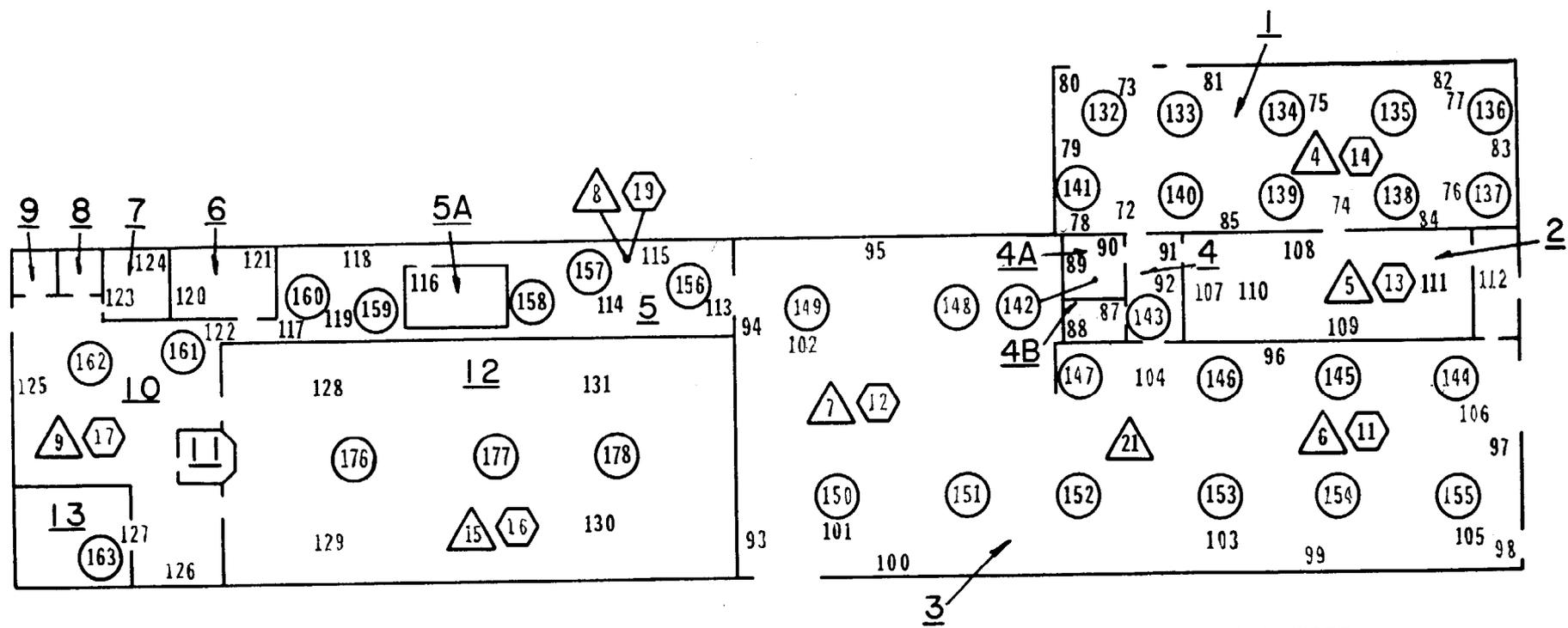
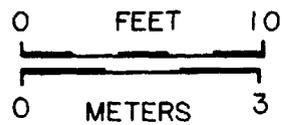
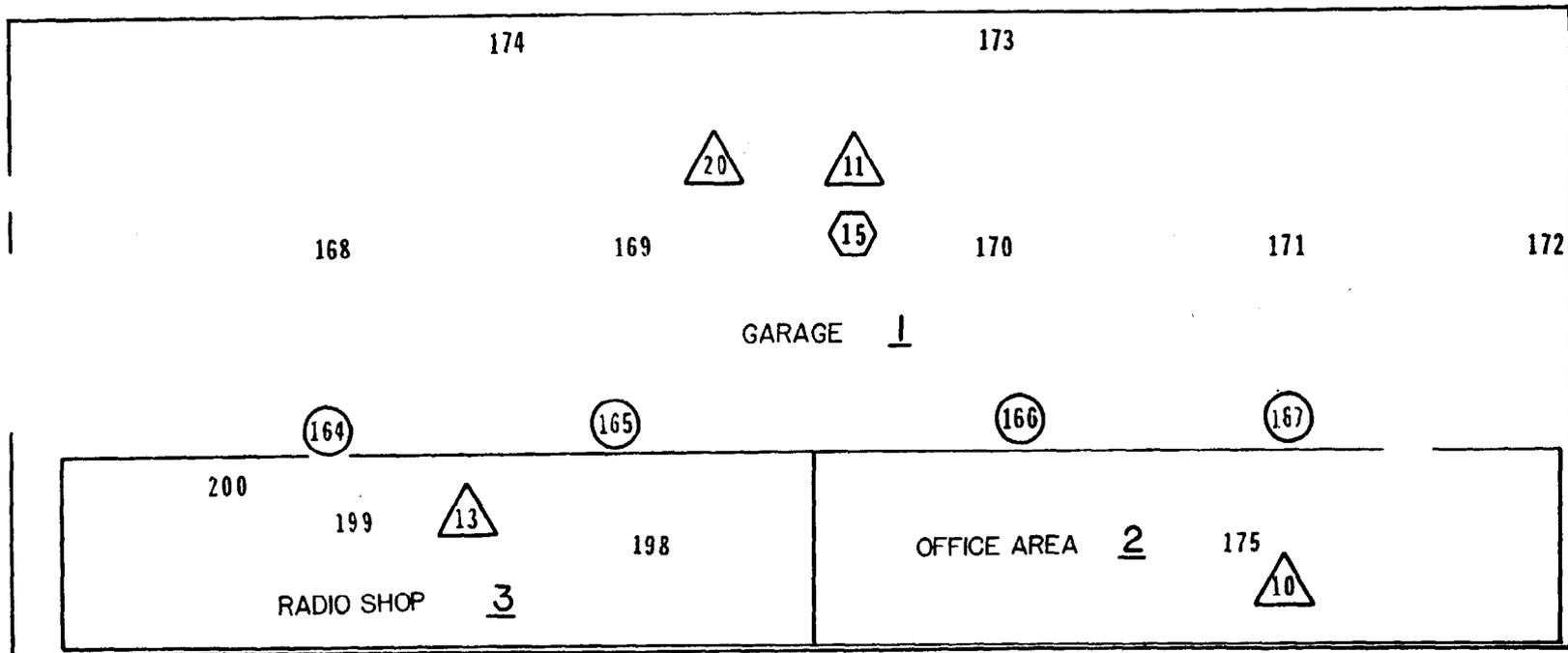


Fig. 8. Soil Coring Locations.
ANL-HP Dwg. No. 82-9



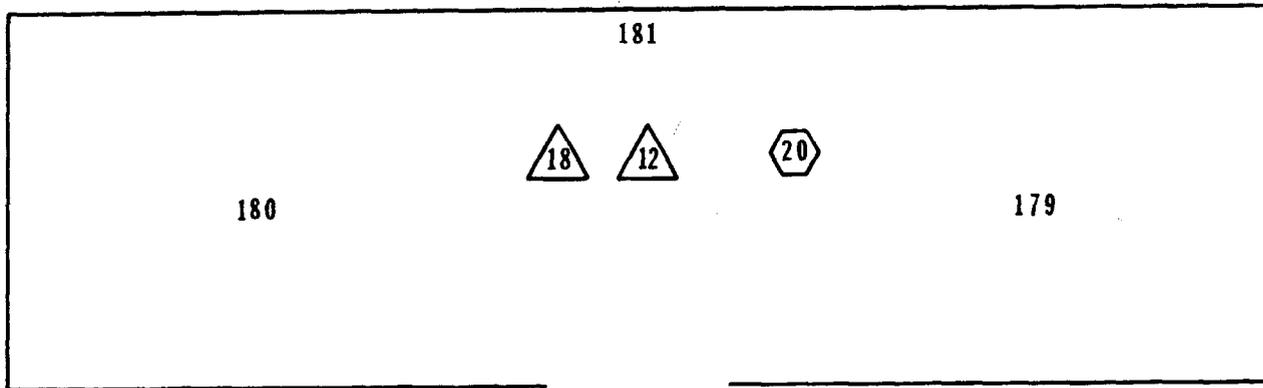
- n ROOM NUMBER
- n SMEAR
- n OVERHEAD SMEAR
- n AIR SAMPLE
- n GAMMA BACKGROUND

Fig. 10. Building 235.
ANL-HP Dwg. No. 82-12.



- n ROOM NUMBER
- n SMEAR
- (n) OVERHEAD SMEAR
- △n AIR SAMPLE
- ⬡n GAMMA BACKGROUND

Fig. 11. Building 236.
ANL-HP Dwg. No. 82-14



- n SMEAR
- △ n AIR SAMPLE
- ⬡ n GAMMA BACKGROUND

Fig. 12. Building 237.
ANL-HP Dwg. No. 82-15

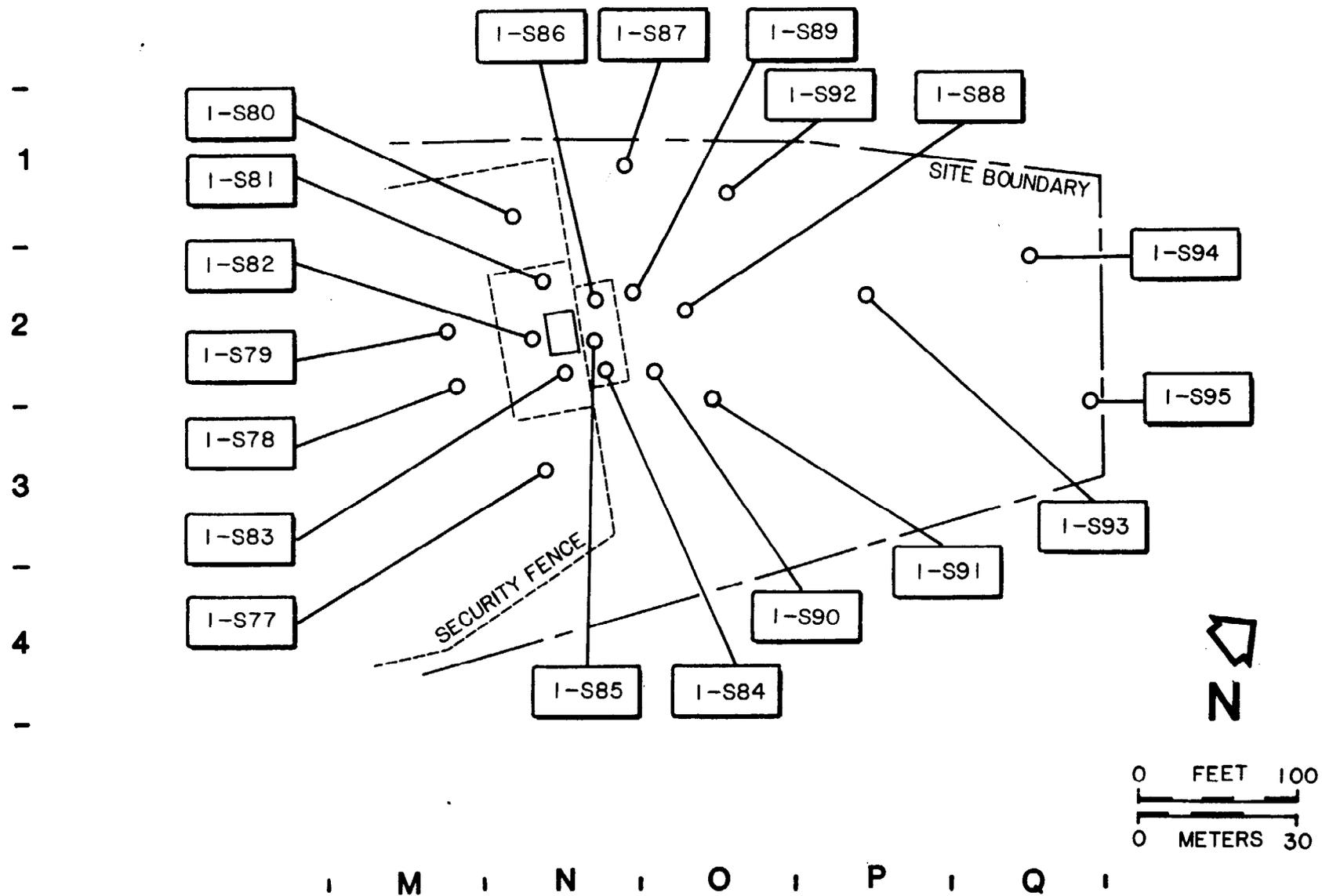


Fig. 13. Soil Boring Locations.
ANL-HP Dwg. No. 82-11

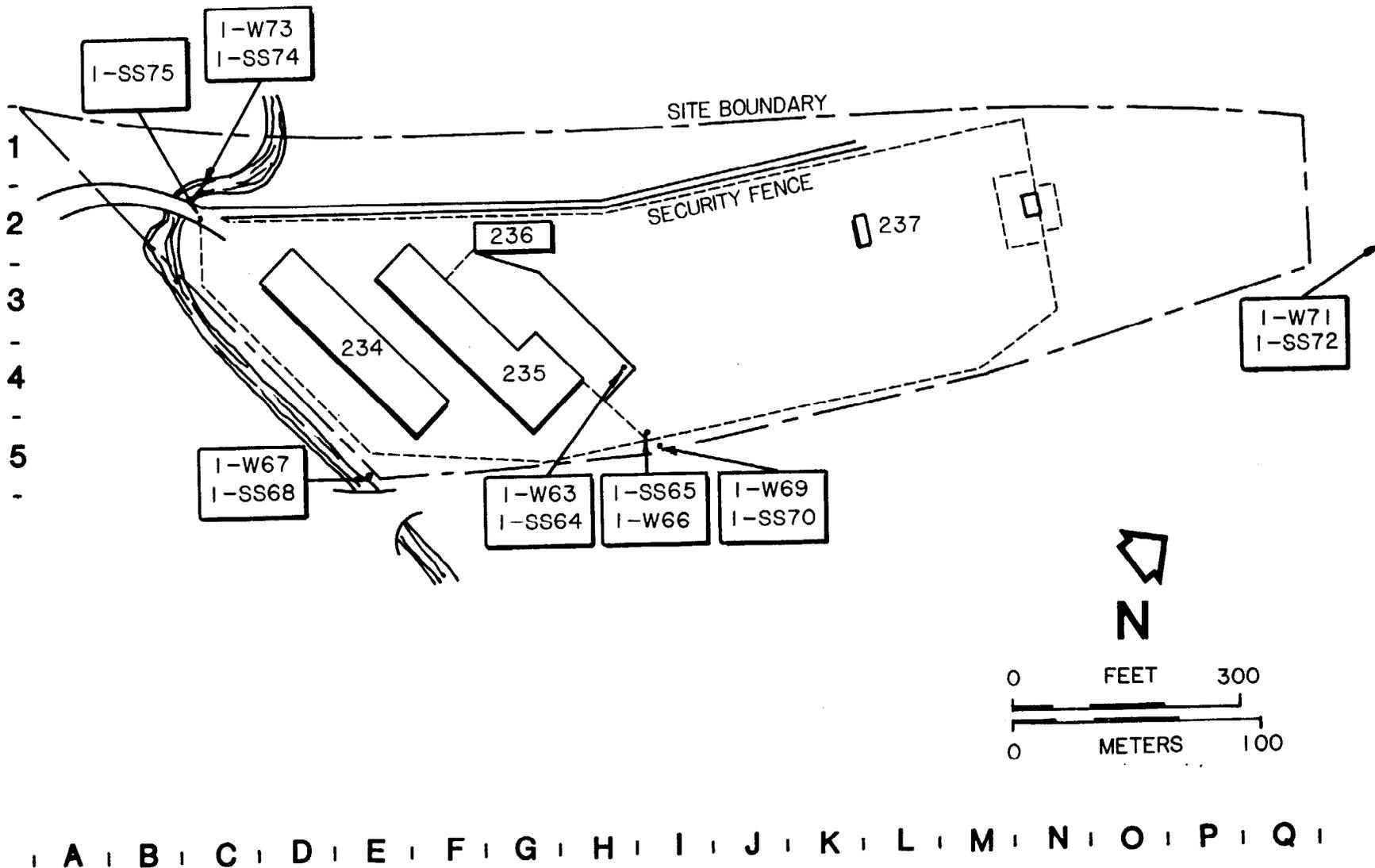


Fig. 14. Water and Sludge Sampling Locations.
ANL-HP Dwg. No. 82-51

TABLE I
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 cm ²)		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 cm ²)	Comments
	Floor	Wall		Beta	Alpha					
AREA I:										
34				1000	BKGD	0.8	40	45000		< 100 cm ²
35				2500	BKGD	0.03	40	50000		< 100 cm ²
36				400	BKGD	0.03	30	20000		< 100 cm ²
37				500	BKGD	0.07	10	10000		< 100 cm ²
38				500	BKGD	0.04	20	3000		< 100 cm ²
39 ^b				15000	250	0.2	20	10000		< 100 cm ²
40 ^b				50000	400	5.0	20	35000		< 100 cm ²
41				2000	BKGD	0.1	20	5000		< 100 cm ²
42				500	BKGD	0.1	20	5000		< 100 cm ²
43				3000	BKGD	0.15	20	15000		< 100 cm ²
44				3000	BKGD	0.5	20	15000		< 100 cm ²
45				2000	100	0.6	20	10000		< 100 cm ²
46				500	BKGD	0.05	20	5000		< 100 cm ²
AREA II: ^c										
47-53, N-1				4500	BKGD	0.14	30	30000		< 700 cm ²
54 ^b , 60, N-3				45000	250	7.0	30	50000		< 200 cm ²

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 cm ²)		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 cm ²)	Comments
	Floor	Wall		Beta	Alpha					
55-57, N-3				4000	BKGD	0.7	30	30000	1100 cm ²	
59, 61-63, N-3				25000	300	0.7	30	50000	47000 cm ²	
58, M-3				400	BKGD	0.03	30	2000	900 cm ²	
71, L-2				300	BKGD	0.03	30	2000	3700 cm ²	
64-66, L-3				1000	BKGD	0.08	30	10000	< 300 cm ²	
67-70, K-3				10000	BKGD	0.02	30	50000	< 400 cm ²	
AREA III General				1000	BKGD	BKGD	10	2000		
AREA IV General				500	BKGD	BKGD	20	3000		
AREA V 72, 73, 77, N-1				250	BKGD	0.03	20	10000	< 300 cm ²	

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 cm ²)		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 cm ²)	Comments
	Floor	Wall		Beta	Alpha					
AREA V (cont'd.)										
74-76, N-2				15000	BKGD	3.0	20	25000		< 300 cm ²
79-82, O-1				250	BKGD	0.04	20	3000		< 300 cm ²
78, O-2				250	BKGD	0.04	20	2000		< 100 cm ²
83, P-1				250	BKGD	0.03	20	2000		< 100 cm ²
89, P-2				250	BKGD	0.03	20	3000		< 100 cm ²
84, Q-1				250	BKGD	0.03	20	2000		< 100 cm ²
85-88, Q-2				400	BKGD	0.03	20	7000		< 300 cm ²

^a All instrument readings include backgrounds (PAC-4GB, 250 cpm, PRM-5-3, 400 cpm, GM, 0.03 mR, PRM-7, 10 μR).

^b Metal chips were found at these locations.

^c Grid locations are shown in Figure 6.

TABLE 2

AMBIENT RADIATION LEVELS

Location ^a			Duration of Measurement (h)	Radiation Level RSS-111 (μ R/h)	Corresponding PRM-7 Measurement (μ R/h)
Area	No.	Grid			
INTERIOR					
Bldg. 234	18		65.5	8.9	7
Bldg. 235	11		4.0	8.8	7
	12		3.0	8.3	7
	13		4.0	9.0	8
	14		3.0	10.2	8
	16		3.4	8.9	8
	17		4.1	9.4	8
	19		6.0	11.2	8
Bldg. 236	15		6.2	8.6	8
Bldg. 237	20		4.1	8.8	8

EXTERIOR					
I	1	M-2	3.3	17.7	34
	2	M-2	2.5	11.6	13
II	3	M-3	4.5	9.8	7
	4	K-3	2.1	11.7	10
	7	H-2	6.2	7.5	5
III	5	H-3	3.0	10.7	9
	6	F-4	3.8	12.6	11
	10	E-4	2.3	10.0	10
IV	9	I-1	2.5	10.2	10
V	8	O-2	1.8	9.6	10
	21	Q-2	2.9	14.6	15
	22	Q-1	3.1	12.4	12

^aLocations are shown on maps in Figures 5 and 9 through 12.

TABLE 3

RADON-CONCENTRATION DETERMINATIONS^a

Air Sample Number	Location	WL ^b	Radon (²²² Rn), pCi/ℓ
4	Building 235, Room 1	0.0074	0.74
5	Building 235, Room 2	0.0065	0.65
6	Building 235, Room 3 East	0.0102	1.02
7	Building 235, Room 3 West	0.0085	0.85
8	Building 235, Room 5	0.0038	0.38
9	Building 235, Room 10	0.0058	0.58
10	Building 236, Office	0.0046	0.46
11	Building 236, Garage	0.0015	0.15
12	Building 237, Shed	0.0004	0.04
13	Building 236, Radio Shop	0.0006	0.06
14	Building 234, East	0.0010	0.10
15	Building 235, Pistol Range	0.0096	0.96
16	Building 234, Center	0.0027	0.27
17	Building 234, West	0.0054	0.54

TABLE 3
(cont'd.)

Air Sample Number	Location	WL ^b
18 ^c	Building 237, Shed	No actinon
20 ^c	Building 236, Garage	No actinon
21 ^c	Building 235, Room 3 Center	No actinon
22 ^c	Building 234, Center	No actinon

^aLocations are shown on maps in Figures 9 through 12.

^bA Working Level (WL) is defined in 10 CFR 712 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 ^{222}Rn per liter of air.

^cActinon determination.

TABLE 4

GAMMA SPECTRAL AND URANIUM FLUOROMETRIC ANALYSES OF SOIL SAMPLES

Sample No.	Gamma Spectra, pCi/g \pm σ				
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric ^a μg/g \pm 10% pCi/g \pm 10%	
<u>SOIL CORINGS</u>					
1-S47-A 2" 5cm	1.54 \pm 0.11	1.03 \pm 0.07	1.36 \pm 0.09	824.0 \pm 82.4	334.0 \pm 33.4 ^b
B 2" 10cm	1.96 \pm 0.14	1.04 \pm 0.07	1.21 \pm 0.09	2227.0 \pm 222.7	902.0 \pm 90.2 ^b
C 2" 15cm	0.53 \pm 0.05	0.86 \pm 0.07	1.52 \pm 0.11	736.0 \pm 73.6	298.0 \pm 29.8 ^b
D 6" 45cm	<0.03	0.35 \pm 0.07	0.62 \pm 0.06	46.2 \pm 4.6	18.7 \pm 1.9 ^b
1-S48-A	1.10 \pm 0.08	1.03 \pm 0.07	1.60 \pm 0.11	595.0 \pm 59.5	241.0 \pm 24.1 ^b
B	0.79 \pm 0.07	1.33 \pm 0.09	1.89 \pm 0.13	484.0 \pm 48.4	196.0 \pm 19.6 ^b
C	0.24 \pm 0.04	1.24 \pm 0.09	1.81 \pm 0.13	100.0 \pm 10.0	40.5 \pm 4.1 ^b
D	0.08 \pm 0.03	0.98 \pm 0.07	1.53 \pm 0.11	47.0 \pm 4.7	19.0 \pm 1.9 ^b
1-S49-A	1.08 \pm 0.11	1.48 \pm 0.15	1.21 \pm 0.12	372.0 \pm 37.2	151.0 \pm 15.1 ^b
B	0.90 \pm 0.09	1.62 \pm 0.16	0.92 \pm 0.09	412.0 \pm 41.2	167.0 \pm 16.7 ^b
C	0.51 \pm 0.05	1.05 \pm 0.11	1.30 \pm 0.13	218.0 \pm 21.8	88.3 \pm 8.8 ^b
D	0.29 \pm 0.03	1.08 \pm 0.11	1.04 \pm 0.10	84.3 \pm 8.4	34.1 \pm 3.4 ^b
1-S50-A	1.03 \pm 0.10	0.67 \pm 0.07	0.83 \pm 0.08	403.0 \pm 40.3	163.0 \pm 16.3 ^b
B	0.67 \pm 0.07	0.83 \pm 0.08	0.97 \pm 0.10	761.0 \pm 76.1	308.0 \pm 30.8 ^b
C	0.53 \pm 0.05	0.73 \pm 0.07	1.03 \pm 0.10	859.0 \pm 85.9	348.0 \pm 34.8 ^b
D	0.63 \pm 0.06	0.87 \pm 0.09	1.40 \pm 0.14	637.0 \pm 63.7	258.0 \pm 25.8 ^b
1-S51-A	0.33 \pm 0.03	0.70 \pm 0.07	1.08 \pm 0.11	2.4 \pm 0.2	1.6 \pm 0.2
B	0.70 \pm 0.07	0.78 \pm 0.08	1.00 \pm 0.10	3.3 \pm 0.3	2.3 \pm 0.2
C	1.01 \pm 0.10	1.01 \pm 0.10	1.18 \pm 0.12	3.8 \pm 0.4	2.6 \pm 0.2
D	0.35 \pm 0.04	0.84 \pm 0.08	0.97 \pm 0.10	2.2 \pm 0.2	1.5 \pm 0.2
1-S52-A	1.12 \pm 0.11	1.25 \pm 0.13	1.85 \pm 0.19	6.4 \pm 0.6	4.4 \pm 0.4
B	0.81 \pm 0.08	1.36 \pm 0.14	1.84 \pm 0.18	5.2 \pm 0.5	3.6 \pm 0.4
C	0.58 \pm 0.06	1.22 \pm 0.12	1.76 \pm 0.18	4.4 \pm 0.4	3.0 \pm 0.3
D	0.04 \pm 0.02	1.01 \pm 0.10	0.90 \pm 0.09	1.9 \pm 0.2	1.3 \pm 0.2

Table 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma$				
	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric	
				$\mu\text{g/g} \pm 10\%$	pCi/g $\pm 10\%$
1-S53-A	1.91 \pm 0.19	0.76 \pm 0.08	1.35 \pm 0.14	7.0 \pm 0.7	3.9 \pm 0.4 ^c
B	2.11 \pm 0.21	1.20 \pm 0.12	2.02 \pm 0.20	5.7 \pm 0.6	3.9 \pm 0.4
C	1.42 \pm 0.14	1.15 \pm 0.12	1.60 \pm 0.16	3.7 \pm 0.4	2.5 \pm 0.3
D	0.20 \pm 0.04	1.26 \pm 0.13	1.61 \pm 0.16	2.4 \pm 0.2	1.6 \pm 0.2
1-S54-A	2.49 \pm 0.25	0.67 \pm 0.07	1.10 \pm 0.11	10.6 \pm 1.1	5.2 \pm 0.5 ^c
B	2.68 \pm 0.27	0.96 \pm 0.10	1.20 \pm 0.12	6.6 \pm 0.7	3.6 \pm 0.4 ^c
C	0.59 \pm 0.06	0.94 \pm 0.09	1.12 \pm 0.11	2.9 \pm 0.3	2.0 \pm 0.2
D	0.12 \pm 0.04	1.44 \pm 0.14	1.15 \pm 0.12	2.1 \pm 0.2	1.4 \pm 0.2
1-S55-A	1.55 \pm 0.16	1.10 \pm 0.11	0.49 \pm 0.05	1.8 \pm 0.2	1.2 \pm 0.2
B	1.74 \pm 0.17	1.20 \pm 0.12	0.88 \pm 0.09	2.6 \pm 0.3	1.8 \pm 0.2
C	1.62 \pm 0.16	1.00 \pm 0.10	1.31 \pm 0.13	4.4 \pm 0.4	3.0 \pm 0.3
D	0.36 \pm 0.04	1.51 \pm 0.15	1.31 \pm 0.13	2.2 \pm 0.2	1.5 \pm 0.2
1-S56-A	0.20 \pm 0.04	<0.06	0.15 \pm 0.05	<1.0	<0.7
B	0.03 \pm 0.04	0.17 \pm 0.05	<0.02	<1.0	<0.7
C	0.21 \pm 0.04	0.34 \pm 0.06	0.53 \pm 0.05	<1.0	<0.7
D	0.13 \pm 0.04	1.05 \pm 0.11	1.35 \pm 0.14	2.0 \pm 0.2	1.4 \pm 0.2
1-S57-A	5.01 \pm 0.50	<0.06	0.91 \pm 0.09	2.1 \pm 0.2	1.4 \pm 0.2
B	7.80 \pm 0.78	0.80 \pm 0.08	1.15 \pm 0.12	1.8 \pm 0.2	1.2 \pm 0.2
C	2.35 \pm 0.24	1.07 \pm 0.11	1.12 \pm 0.11	1.7 \pm 0.2	1.2 \pm 0.2
D	0.99 \pm 0.10	0.70 \pm 0.07	1.20 \pm 0.12	2.1 \pm 0.2	1.4 \pm 0.2
1-S58-A	1.44 \pm 0.14	1.38 \pm 0.14	0.80 \pm 0.08	1.9 \pm 0.2	1.3 \pm 0.2
B	1.65 \pm 0.17	0.94 \pm 0.09	0.88 \pm 0.09	1.9 \pm 0.2	1.3 \pm 0.2
C	0.93 \pm 0.09	1.10 \pm 0.11	1.24 \pm 0.12	1.8 \pm 0.2	1.2 \pm 0.2
D	0.45 \pm 0.05	1.06 \pm 0.11	1.01 \pm 0.10	1.7 \pm 0.2	1.2 \pm 0.2

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g ± σ				
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric µg/g ± 10%	pCi/g ± 10%
1-S59-A	1.55 ± 0.16	1.31 ± 0.13	1.86 ± 0.19	4.2 ± 0.4	2.9 ± 0.3
B	2.74 ± 0.27	1.18 ± 0.12	1.72 ± 0.17	4.8 ± 0.5	3.3 ± 0.3
C	2.27 ± 0.23	0.68 ± 0.07	1.46 ± 0.15	3.6 ± 0.4	2.5 ± 0.3
D	0.33 ± 0.04	1.11 ± 0.11	2.01 ± 0.20	3.6 ± 0.4	2.5 ± 0.3
1-S60-A	5.19 ± 0.52	1.47 ± 0.15	2.07 ± 0.21	3.9 ± 0.4	2.7 ± 0.3
B	3.48 ± 0.35	2.19 ± 0.22	3.39 ± 0.34	6.9 ± 0.7	4.7 ± 0.5
C	0.78 ± 0.08	3.32 ± 0.33	4.85 ± 0.49	8.9 ± 0.9	6.1 ± 0.6
D	0.20 ± 0.04	1.09 ± 0.11	1.45 ± 0.15	2.5 ± 0.3	1.7 ± 0.2
1-S61-A	0.97 ± 0.10	1.15 ± 0.12	0.81 ± 0.08	2.3 ± 0.2	1.6 ± 0.2
B	0.62 ± 0.06	1.39 ± 0.14	0.73 ± 0.07	2.5 ± 0.3	1.7 ± 0.2
C	0.61 ± 0.06	0.88 ± 0.09	0.76 ± 0.08	2.6 ± 0.3	1.8 ± 0.2
D	0.95 ± 0.10	1.03 ± 0.16	0.84 ± 0.08	3.0 ± 0.3	2.1 ± 0.2
1-S62-A	1.10 ± 0.11	1.59 ± 0.16	1.10 ± 0.11	27.0 ± 2.7	11.8 ± 1.2 ^c
B	0.50 ± 0.05	0.80 ± 0.08	1.67 ± 0.17	15.5 ± 1.6	7.6 ± 0.8 ^c
C	0.11 ± 0.03	1.28 ± 0.13	1.20 ± 0.12	4.6 ± 0.5	3.2 ± 0.3
D	<0.03	0.90 ± 0.09	1.21 ± 0.12	2.5 ± 0.3	1.7 ± 0.2
1-S75 ^d	1.63 ± 0.16	0.07 ± 0.04	1.34 ± 0.13	1.9 ± 0.2	1.3 ± 0.2
1-S76 ^e	0.48 ± 0.05	1.75 ± 0.18	3.84 ± 0.38	962.0 ± 96.0	390.0 ± 39.0 ^b
<u>SOIL BORINGS</u>					
1-S77-1	0.23 ± 0.04	0.58 ± 0.06	0.72 ± 0.07	3.5 ± 0.4	2.0 ± 0.2 ^c
2	0.30 ± 0.04	0.73 ± 0.07	0.43 ± 0.04	3.8 ± 0.4	1.9 ± 0.2 ^c
4	0.34 ± 0.04	0.49 ± 0.05	0.36 ± 0.04	1.1 ± 0.1	0.8 ± 0.1
5	0.51 ± 0.05	1.21 ± 0.12	0.78 ± 0.08	2.6 ± 0.3	1.8 ± 0.2
6	0.17 ± 0.04	0.83 ± 0.08	0.74 ± 0.07	2.6 ± 0.3	1.8 ± 0.2

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g±σ					
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric		
				µg/g ± 10%	pCi/g ± 10%	
1-S78-1	0.35 ± 0.04	2.29 ± 0.23	2.42 ± 0.24	11.2 ± 1.1	6.5 ± 0.7 ^c	
2	0.57 ± 0.06	1.14 ± 0.11	1.11 ± 0.11	8.3 ± 0.8	4.3 ± 0.4 ^c	
3	0.57 ± 0.06	3.66 ± 0.37	2.32 ± 0.23	7.5 ± 0.8	5.1 ± 0.5	
4	0.21 ± 0.03	1.98 ± 0.20	1.69 ± 0.17	3.0 ± 0.3	2.0 ± 0.2	
5	0.24 ± 0.03	1.76 ± 0.18	1.76 ± 0.18	3.2 ± 0.3	2.2 ± 0.2	
6	0.25 ± 0.04	1.82 ± 0.18	1.69 ± 0.17	3.6 ± 0.4	2.5 ± 0.3	
1-S79-1	0.53 ± 0.05	1.65 ± 0.17	1.78 ± 0.18	22.5 ± 2.3	10.6 ± 1.1 ^c	
2	0.44 ± 0.04	1.29 ± 0.13	1.15 ± 0.12	4.9 ± 0.5	2.9 ± 0.3 ^c	
3	0.18 ± 0.04	0.15 ± 0.05	0.29 ± 0.03	0.9 ± 0.1	0.6 ± 0.1	
4	1.70 ± 0.17	0.33 ± 0.07	0.68 ± 0.07	1.3 ± 0.1	0.9 ± 0.1	
5	<0.03	0.23 ± 0.04	0.60 ± 0.06	2.0 ± 0.2	1.4 ± 0.1	
6	<0.03	1.15 ± 0.12	0.69 ± 0.07	1.6 ± 0.2	1.1 ± 0.1	
1-S80-1	0.36 ± 0.04	0.89 ± 0.09	1.07 ± 0.11	23.3 ± 2.3	10.3 ± 1.0 ^c	
2	<0.03	1.85 ± 0.19	2.10 ± 0.21	6.7 ± 0.7	4.4 ± 0.4 ^c	
3	<0.03	<0.06	0.48 ± 0.05	1.7 ± 0.2	1.2 ± 0.1 ^c	
4	<0.03	0.68 ± 0.07	0.38 ± 0.04	4.1 ± 0.4	2.0 ± 0.2 ^c	
5	1.35 ± 0.14	1.01 ± 0.10	1.48 ± 0.15	7.0 ± 0.7	4.0 ± 0.4 ^c	
6	0.22 ± 0.02	0.98 ± 0.10	0.60 ± 0.06	7.4 ± 0.7	3.5 ± 0.4 ^c	
1-S81-1	0.20 ± 0.04	0.95 ± 0.10	1.41 ± 0.14	622.0 ± 62.2	252.0 ± 25.2 ^b	
2	0.77 ± 0.08	0.24 ± 0.02	1.36 ± 0.14	1400.0 ± 140.0	567.0 ± 56.7 ^b	
4	0.22 ± 0.04	0.60 ± 0.06	1.44 ± 0.14	232.0 ± 23.2	94.0 ± 9.4 ^b	
5	0.34 ± 0.03	0.78 ± 0.08	0.83 ± 0.08	101.0 ± 10.1	40.9 ± 4.1 ^b	
6	<0.03	0.60 ± 0.06	1.19 ± 0.12	14.2 ± 1.4	6.7 ± 0.7 ^b	
1-S82-1	0.57 ± 0.06	1.32 ± 0.13	1.06 ± 0.11	119.0 ± 12.0	48.2 ± 4.8 ^b	
2	0.51 ± 0.05	1.42 ± 0.14	1.15 ± 0.12	62.4 ± 6.2	25.3 ± 2.5 ^b	
3	0.88 ± 0.09	1.18 ± 0.12	1.17 ± 0.12	54.3 ± 5.4	22.0 ± 2.2 ^b	
4	0.21 ± 0.02	1.35 ± 0.14	0.74 ± 0.07	10.4 ± 1.0	4.8 ± 0.5 ^c	
5	0.21 ± 0.02	0.25 ± 0.05	0.36 ± 0.04	2.0 ± 0.2	1.4 ± 0.2	
6	0.25 ± 0.02	0.63 ± 0.06	0.79 ± 0.08	1.6 ± 0.2	1.1 ± 0.1	

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g ± σ				
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric μg/g ± 10%	pCi/g ± 10%
1-S83-1	0.08 ± 0.05	0.60 ± 0.06	1.13 ± 0.11	136.0 ± 13.6	55.1 ± 5.5 ^b
2	0.37 ± 0.04	0.61 ± 0.06	1.56 ± 0.16	928.0 ± 92.8	376.0 ± 37.6 ^b
3	<0.03	0.31 ± 0.09	0.60 ± 0.06	33.4 ± 3.4	13.5 ± 1.4 ^c
4	<0.03	0.34 ± 0.07	0.83 ± 0.08	15.2 ± 1.5	6.8 ± 0.7 ^b
5	<0.03	0.13 ± 0.05	1.04 ± 0.10	39.0 ± 3.9	15.8 ± 1.6 ^b
6	0.08 ± 0.05	0.46 ± 0.07	0.99 ± 0.10	35.9 ± 3.6	14.5 ± 1.5 ^b
1-S84-1	0.90 ± 0.09	0.89 ± 0.09	1.06 ± 0.11	257.0 ± 26.0	104.0 ± 10.4 ^b
2	0.35 ± 0.04	1.07 ± 0.01	1.35 ± 0.14	136.0 ± 14.0	55.1 ± 5.5 ^b
3	0.63 ± 0.06	1.14 ± 0.11	0.28 ± 0.03	8.9 ± 0.9	3.6 ± 0.4 ^b
5	0.13 ± 0.03	0.73 ± 0.07	0.91 ± 0.09	6.1 ± 0.6	3.2 ± 0.3 ^c
6	<0.03	0.58 ± 0.09	1.01 ± 0.10	5.8 ± 0.6	3.2 ± 0.3 ^{c4}
1-S85-1	3.62 ± 0.4	0.94 ± 0.09	0.68 ± 0.07	710.0 ± 71.0	288.0 ± 28.8 ^b
2	0.68 ± 0.07	1.75 ± 0.18	1.62 ± 0.16	1452.0 ± 145.2	588.0 ± 58.8 ^b
3	0.47 ± 0.05	3.81 ± 0.38	2.42 ± 0.24	254.0 ± 25.4	70.1 ± 7.0 ^b
4	0.23 ± 0.02	0.40 ± 0.08	1.00 ± 0.10	27.4 ± 2.7	11.1 ± 1.1 ^b
5	0.38 ± 0.04	1.46 ± 0.15	1.07 ± 0.11	18.9 ± 1.9	7.6 ± 0.8 ^b
6	0.47 ± 0.05	0.62 ± 0.06	0.78 ± 0.08	10.1 ± 1.0	4.1 ± 0.4 ^b
1-S86-1	0.57 ± 0.06	0.36 ± 0.07	0.78 ± 0.08	393.0 ± 39.3	159.0 ± 15.9 ^b
2	1.18 ± 0.12	0.57 ± 0.09	1.41 ± 0.14	597.0 ± 59.7	242.0 ± 24.2 ^b
3	0.55 ± 0.06	1.12 ± 0.11	1.04 ± 0.10	33.5 ± 3.4	13.6 ± 1.4 ^b
4	<0.03	1.18 ± 0.12	1.25 ± 0.12	19.9 ± 2.0	8.1 ± 0.8 ^b
5	0.03 ± 0.02	0.84 ± 0.08	1.30 ± 0.13	8.2 ± 0.8	4.4 ± 0.4 ^b
6	<0.03	0.56 ± 0.08	0.90 ± 0.09	2.4 ± 0.2	1.6 ± 0.2
1-S87-1	1.00 ± 0.10	1.74 ± 0.17	11.02 ± 1.10	14.1 ± 1.4	9.6 ± 1.0
2	0.19 ± 0.02	1.03 ± 0.10	1.72 ± 0.17	2.4 ± 0.2	1.6 ± 0.2
3	0.42 ± 0.04	1.56 ± 0.16	2.11 ± 0.21	4.5 ± 0.5	3.1 ± 0.3
4	0.15 ± 0.04	2.21 ± 0.22	2.32 ± 0.23	5.7 ± 0.6	3.9 ± 0.4
5	0.15 ± 0.04	2.15 ± 0.22	2.44 ± 0.24	5.8 ± 0.6	4.0 ± 0.4
6	0.13 ± 0.03	2.35 ± 0.24	2.05 ± 0.20	4.4 ± 0.4	3.0 ± 0.3

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma$				
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric $\mu\text{g/g} \pm 10\%$	Uranium Fluorometric pCi/g $\pm 10\%$
1-S88-1	0.76 \pm 0.08	0.65 \pm 0.06	1.03 \pm 0.10	9.0 \pm 0.9	4.5 \pm 0.4 ^c
2	0.29 \pm 0.04	1.40 \pm 0.14	1.06 \pm 0.11	4.8 \pm 0.5	2.8 \pm 0.3 ^c
3	1.24 \pm 0.12	0.33 \pm 0.10	1.21 \pm 0.12	3.0 \pm 0.3	2.0 \pm 0.2
4	0.36 \pm 0.04	0.98 \pm 0.10	0.93 \pm 0.09	2.9 \pm 0.3	2.0 \pm 0.2
5	0.61 \pm 0.06	2.04 \pm 0.20	1.77 \pm 0.18	3.8 \pm 0.4	2.6 \pm 0.3
6	1.19 \pm 0.12	2.18 \pm 0.22	2.65 \pm 0.26	3.7 \pm 0.4	2.5 \pm 0.3
1-S89-1	0.15 \pm 0.04	0.57 \pm 0.09	0.56 \pm 0.06	8.7 \pm 0.9	4.0 \pm 0.4 ^c
2	0.16 \pm 0.04	0.08 \pm 0.04	0.40 \pm 0.04	5.5 \pm 0.6	2.6 \pm 0.3 ^c
3	<0.03	0.48 \pm 0.07	0.89 \pm 0.09	3.2 \pm 0.3	2.2 \pm 0.2
4	0.11 \pm 0.03	0.50 \pm 0.08	0.89 \pm 0.09	2.1 \pm 0.2	1.4 \pm 0.1
5	0.33 \pm 0.04	0.67 \pm 0.07	0.75 \pm 0.08	1.8 \pm 0.2	1.2 \pm 0.1
6	0.33 \pm 0.04	0.83 \pm 0.08	0.99 \pm 0.10	2.4 \pm 0.2	1.6 \pm 0.2
1-S90-1	1.50 \pm 0.15	1.14 \pm 0.11	1.07 \pm 0.11	42.3 \pm 4.2	17.1 \pm 1.7 ^b
2	0.21 \pm 0.04	0.70 \pm 0.07	0.82 \pm 0.08	6.6 \pm 0.7	3.3 \pm 0.3 ^c
3	0.15 \pm 0.04	0.47 \pm 0.09	0.73 \pm 0.07	3.7 \pm 0.4	2.1 \pm 0.2 ^c
4	<0.03	0.71 \pm 0.07	0.75 \pm 0.08	1.4 \pm 0.2	1.0 \pm 0.1
5	0.14 \pm 0.04	1.12 \pm 0.11	0.85 \pm 0.08	1.8 \pm 0.2	1.2 \pm 0.1
6	0.06 \pm 0.02	<0.06	0.78 \pm 0.08	1.7 \pm 0.2	1.2 \pm 0.1
1-S91-1	0.44 \pm 0.04	2.96 \pm 0.30	3.61 \pm 0.36	11.6 \pm 1.2	7.9 \pm 0.8
2	0.03 \pm 0.02	1.13 \pm 0.11	0.91 \pm 0.09	2.5 \pm 0.3	1.7 \pm 0.2
3	<0.03	1.43 \pm 0.14	1.29 \pm 0.13	2.8 \pm 0.3	1.9 \pm 0.2
4	<0.03	2.21 \pm 0.22	1.70 \pm 0.17	3.7 \pm 0.4	2.5 \pm 0.3
5	<0.03	0.43 \pm 0.06	0.39 \pm 0.04	2.2 \pm 0.2	1.5 \pm 0.2
6	<0.03	1.72 \pm 0.17	1.02 \pm 0.10	4.4 \pm 0.5	3.0 \pm 0.3

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g to				
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric µg/g ± 10%	pCi/g ± 10%
1-S92-1	BDL ^g	2.84 ± 0.28	13.66 ± 1.37	1.4 ± 0.2	1.0 ± 0.1
2	0.19 ± 0.06	2.08 ± 0.21	11.67 ± 1.17	2.5 ± 0.3	1.7 ± 0.2
3	0.14 ± 0.04	0.94 ± 0.09	4.44 ± 0.44	2.3 ± 0.2	1.6 ± 0.2
4	0.09 ± 0.04	0.93 ± 0.09	1.98 ± 0.20	1.9 ± 0.2	1.3 ± 0.1
5	BDL	0.35 ± 0.07	0.97 ± 0.10	1.0 ± 0.1	0.7 ± 0.1
6	BDL	0.55 ± 0.08	0.87 ± 0.09	1.2 ± 0.1	0.8 ± 0.1
1-S93-1	0.25 ± 0.04	0.79 ± 0.08	1.65 ± 0.16	2.7 ± 0.3	1.8 ± 0.2
2	0.22 ± 0.04	0.92 ± 0.09	1.34 ± 0.13	2.0 ± 0.2	1.4 ± 0.2
3	0.04 ± 0.02	1.10 ± 0.11	1.32 ± 0.13	2.3 ± 0.2	1.6 ± 0.2
4	<0.03	0.83 ± 0.08	0.95 ± 0.10	2.9 ± 0.3	2.0 ± 0.2
5	<0.03	1.06 ± 0.11	1.72 ± 0.17	3.3 ± 0.3	2.3 ± 0.2
6	0.03 ± 0.02	0.95 ± 0.10	1.28 ± 0.13	2.7 ± 0.3	1.8 ± 0.2
1-S94-1	0.63 ± 0.06	2.08 ± 0.21	5.54 ± 0.55	17.8 ± 1.8	12.2 ± 1.2
2	0.03 ± 0.02	1.63 ± 0.16	7.99 ± 0.80	10.3 ± 1.0	7.0 ± 0.7
3	0.17 ± 0.05	1.15 ± 0.12	3.44 ± 0.34	14.1 ± 1.4	9.6 ± 1.0
4	<0.03	-	1.46 ± 0.15	2.4 ± 0.2	1.6 ± 0.2
5	<0.03	0.77 ± 0.08	1.23 ± 0.12	1.8 ± 0.2	1.2 ± 0.1
6	<0.03	0.58 ± 0.09	1.85 ± 0.18	2.4 ± 0.2	1.6 ± 0.2
1-S95-1	0.55 ± 0.06	2.18 ± 0.22	5.04 ± 0.50	2.6 ± 0.3	1.8 ± 0.2
2	0.12 ± 0.04	1.88 ± 0.19	4.24 ± 0.42	2.9 ± 0.3	2.0 ± 0.7
3	0.28 ± 0.04	1.76 ± 0.18	4.14 ± 0.41	2.3 ± 0.2	1.6 ± 0.2
4	0.09 ± 0.04	0.97 ± 0.10	2.41 ± 0.24	4.1 ± 0.4	2.8 ± 0.3
5	0.15 ± 0.04	0.31 ± 0.03	2.09 ± 0.21	4.3 ± 0.4	2.9 ± 0.3
6	<0.03	1.00 ± 0.10	4.18 ± 0.42	13.5 ± 1.4	9.2 ± 0.9
1-R96 ^f	BDL	274.0 ± 27.0	215.0 ± 22.0	708.0 ± 71.0	483.0 ± 48

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma$				
	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric $\mu\text{g/g} \pm 10\%$	Uranium Fluorometric $\text{pCi/g} \pm 10\%$
1-S97-A	0.84 \pm 0.08	0.85 \pm 0.08	1.01 \pm 0.10	3.4 \pm 0.3	2.3 \pm 0.2
B	0.24 \pm 0.05	1.11 \pm 0.11	1.07 \pm 0.11	3.4 \pm 0.3	2.3 \pm 0.2
C	0.16 \pm 0.05	0.92 \pm 0.09	1.24 \pm 0.12	2.4 \pm 0.2	1.6 \pm 0.2
D	0.30 \pm 0.04	0.95 \pm 0.10	1.45 \pm 0.14	2.6 \pm 0.3	1.8 \pm 0.2
1-S98-A	1.39 \pm 0.14	0.70 \pm 0.07	0.88 \pm 0.09	3.3 \pm 0.3	2.1 \pm 0.2 ^c
B	0.78 \pm 0.08	0.84 \pm 0.08	0.85 \pm 0.08	6.0 \pm 0.6	3.1 \pm 0.3 ^c
C	0.35 \pm 0.04	0.64 \pm 0.06	0.48 \pm 0.05	7.6 \pm 0.8	3.5 \pm 0.4 ^c
D	0.26 \pm 0.04	1.09 \pm 0.11	0.82 \pm 0.04	7.5 \pm 0.8	3.7 \pm 0.4 ^c
1-S99-A	0.47 \pm 0.05	1.82 \pm 0.18	2.04 \pm 0.20	251.0 \pm 25.1	102.0 \pm 10.2 ^b
B	0.16 \pm 0.05	2.45 \pm 0.24	4.08 \pm 0.41	322.0 \pm 32.2	130.0 \pm 13.0 ^b
C	0.16 \pm 0.05	2.25 \pm 0.22	5.89 \pm 0.59	545.0 \pm 54.5	221.0 \pm 22.1 ^b
D	<0.03	1.16 \pm 0.12	5.42 \pm 0.54	272.0 \pm 27.2	110.0 \pm 11.0 ^b
1-S100-A	1.17 \pm 0.12	1.75 \pm 0.18	2.10 \pm 0.21	5.3 \pm 0.5	3.6 \pm 0.4
B	0.66 \pm 0.07	1.90 \pm 0.19	2.06 \pm 0.21	5.8 \pm 0.6	4.0 \pm 0.4
C	0.60 \pm 0.06	2.22 \pm 0.22	1.95 \pm 0.20	8.8 \pm 0.9	3.6 \pm 0.4 ^c
D	0.69 \pm 0.07	3.70 \pm 0.37	3.09 \pm 0.31	8.3 \pm 0.8	3.4 \pm 0.4 ^c
1-S101-A	0.51 \pm 0.05	0.86 \pm 0.09	0.77 \pm 0.08	4.5 \pm 0.5	3.1 \pm 0.3
B	0.05 \pm 0.02	1.15 \pm 0.12	1.01 \pm 0.10	3.0 \pm 0.3	2.0 \pm 0.2
C	0.21 \pm 0.04	1.24 \pm 0.12	0.90 \pm 0.09	3.8 \pm 0.4	2.6 \pm 0.3
D	0.34 \pm 0.05	1.00 \pm 0.10	1.25 \pm 0.12	3.3 \pm 0.2	2.3 \pm 0.2
1-S102 ^e	1.55 \pm 0.16	1.29 \pm 0.13	3.06 \pm 0.31	12.8 \pm 1.3	7.7 \pm 0.8 ^c
1-S103 ^h -A	3.82 \pm 0.38	0.94 \pm 0.09	1.73 \pm 0.17	1.75x10 ⁴ \pm 1.75x10 ³	7.1x10 ³ \pm 7.1x10 ² ^b
B	1.48 \pm 0.15	-	1.10 \pm 0.11	2.73x10 ³ \pm 2.73x10 ²	1.1x10 ³ \pm 1.1x10 ² ^b
C	0.18 \pm 0.05	1.09 \pm 0.11	1.40 \pm 0.14	463.0 \pm 46.3	188.0 \pm 18.8 ^b
D	0.07 \pm 0.04	1.30 \pm 0.13	1.65 \pm 0.16	343.0 \pm 34.3	139.0 \pm 13.9 ^b
E	<0.03	1.22 \pm 0.12	2.17 \pm 0.22	146.0 \pm 14.6	59.1 \pm 5.9 ^b

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma$				
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	Uranium Fluorometric	
				$\mu\text{g/g} \pm 10\%$	pCi/g $\pm 10\%$
1-S104-A	3.13 \pm 0.31	BDL	1.88 \pm 0.19	3.10x10 ³ \pm 3.1x10 ²	1.3x10 ³ \pm 1.3x10 ² ^b
B	1.01 \pm 0.10	BDL	1.34 \pm 0.13	6.39x10 ⁴ \pm 6.4x10 ³	2.6x10 ⁴ \pm 2.6x10 ³ ^b
C	2.14 \pm 0.21	0.72 \pm 0.07	1.23 \pm 0.12	5.26x10 ³ \pm 5.3x10 ²	2.1x10 ³ \pm 2.1x10 ² ^b
1-S105 ^e	2.53 \pm 0.25	0.30 \pm 0.09	1.45 \pm 0.14	1.67x10 ⁴ \pm 1.7x10 ³	6.8x10 ³ \pm 6.8x10 ² ^b
1-S106 ^e	2.62 \pm 0.26	0.85 \pm 0.08	1.05 \pm 0.10	1.67x10 ⁴ \pm 1.7x10 ³	6.8x10 ³ \pm 6.8x10 ² ^b
BACKGROUND SOIL ENVIRONMENTAL					
1-SB107-A	1.25 \pm 0.06	1.2 \pm 0.1	0.96 \pm 0.06	3.4 \pm 0.5	2.3 \pm 0.3
B				4.3 \pm 0.6	2.9 \pm 0.4
C				4.1 \pm 0.6	2.8 \pm 0.4
D				3.1 \pm 0.4	2.1 \pm 0.3
1-SB108-A	0.91 \pm 0.05	1.5 \pm 0.1	1.00 \pm 0.07	2.5 \pm 0.5	1.7 \pm 0.3
B				6.8 \pm 0.6	4.6 \pm 0.4
C				3.5 \pm 0.5	2.4 \pm 0.3
D				3.2 \pm 0.4	2.2 \pm 0.3
1-SB109-A	2.20 \pm 0.10	0.9 \pm 0.1	0.68 \pm 0.06	2.5 \pm 0.4	1.7 \pm 0.3
B				2.2 \pm 0.4	1.5 \pm 0.3
C				1.3 \pm 0.4	0.9 \pm 0.3
D				2.4 \pm 0.4	1.6 \pm 0.3
1-SB110-A	<0.05	0.91 \pm 0.1	0.72 \pm 0.06	4.0 \pm 0.4	2.7 \pm 0.3
B				17.4 \pm 0.8	11.8 \pm 0.5
C				5.0 \pm 0.4	3.4 \pm 0.3
D				3.2 \pm 0.4	2.2 \pm 0.3

TABLE 4
(cont'd.)

^aConversion per Appendix 5.

^bDepleted uranium conversion factor of 0.406 pCi/g based on abundances given in Table 6.

^cDepleted uranium corrected for natural uranium based on ²²⁶Ra concentration.

^dSoil from sewer (see Fig. 14).

^eSurface scrapings.

^fRock sample from surface (see Fig. 8).

^gBDL = Below Detectable Limits

^hSample subdivided into 2-in, 2-in, 2-in, 6-in, and 1-in segments.

TABLE 5

GAMMA SPECTRAL AND URANIUM FLUOROMETRIC
ANALYSES OF WATER (W) AND SLUDGE (SS) SAMPLES

Sample No.	Volume (ml)	Gamma Spectra (pCi/ml $\pm\sigma$)		Uranium Fluorometric	
		²³² Th Decay Chain	²²⁶ Ra Decay Chain	$\mu\text{g/ml}\pm 10\%$	pCi/ml $\pm 10\%$ ^a
DISSOLVED SOLIDS					
1-W63	132	<0.01	<0.01	0.0011 \pm 0.0001	7.5 $\times 10^{-4}$ \pm 7.5 $\times 10^{-5}$
1-SS64	41	0.36 \pm 0.07	<0.01	0.0027 \pm 0.0003	1.8 $\times 10^{-3}$ \pm 1.8 $\times 10^{-4}$
1-SS65	43	BDL ^b	0.21 \pm 0.03	0.0012 \pm 0.0001	8.2 $\times 10^{-4}$ \pm 8.2 $\times 10^{-5}$
1-W66	147	<0.01	<0.01	0.0014 \pm 0.0001	9.6 $\times 10^{-4}$ \pm 9.6 $\times 10^{-5}$
1-W67	125	0.05 \pm 0.02	<0.01	0.0002 \pm 0.0001	1.4 $\times 10^{-4}$ \pm 1.4 $\times 10^{-5}$
1-SS68	42	BDL	0.09 \pm 0.03	0.0010 \pm 0.0001	6.8 $\times 10^{-4}$ \pm 6.8 $\times 10^{-5}$
1-W69	135	<0.01	<0.01	0.0007 \pm 0.0001	2.8 $\times 10^{-4}$ \pm 2.8 $\times 10^{-5}$ ^c
1-SS70	58	0.05 \pm 0.02	<0.01	0.0022 \pm 0.0002	1.5 $\times 10^{-3}$ \pm 1.5 $\times 10^{-4}$
1-W71	135	<0.01	<0.01	0.0002 \pm 0.0001	1.4 $\times 10^{-4}$ \pm 1.4 $\times 10^{-5}$
1-SS72	85	0.04 \pm 0.01	<0.01	0.0012 \pm 0.0001	8.2 $\times 10^{-4}$ \pm 8.2 $\times 10^{-5}$
1-W73	132	0.12 \pm 0.05	<0.01	0.0008 \pm 0.0001	5.5 $\times 10^{-4}$ \pm 5.5 $\times 10^{-5}$
1-SS74	47	BDL	0.07 \pm 0.02	0.0006 \pm 0.0001	4.1 $\times 10^{-4}$ \pm 4.1 $\times 10^{-5}$
1-W86	135	0.04 \pm 0.02	0.03 \pm 0.02	0.0215 \pm 0.0022	8.7 $\times 10^{-3}$ \pm 8.7 $\times 10^{-4}$ ^c
1-W90	38	BDL	<0.01	0.0011 \pm 0.0001	7.5 $\times 10^{-4}$ \pm 7.5 $\times 10^{-5}$
1-W91	147	BDL	0.02 \pm 0.01	0.0038 \pm 0.0004	1.5 $\times 10^{-3}$ \pm 1.5 $\times 10^{-4}$ ^c
1-W92	137	BDL	<0.01	0.0035 \pm 0.0004	2.4 $\times 10^{-3}$ \pm 2.4 $\times 10^{-4}$
1-W93	134	<0.01	<0.01	0.0014 \pm 0.0002	9.6 $\times 10^{-4}$ \pm 9.6 $\times 10^{-5}$
1-W95	142	0.04 \pm 0.02	<0.01	0.0134 \pm 0.0013	9.2 $\times 10^{-4}$ \pm 9.2 $\times 10^{-5}$

TABLE 5
(cont'd.)

Sample No.	Weight (g)	Gamma Spectra (pCi/ml $\pm\sigma$)		Uranium Fluorometric	
		²³² Th Decay Chain	²²⁶ Ra Decay Chain	$\mu\text{g/g}\pm 10\%$	pCi/g $\pm 10\%$
SUSPENDED SOLIDS					
1-W63	0.033	BDL	<0.02	3.0 \pm 0.3	2.0 \pm 0.2
1-SS64	109.2	1.00 \pm 0.10	<0.02	3.4 \pm 0.3	2.3 \pm 0.2
1-SS65	97.4	0.36 \pm 0.07	1.04 \pm 0.10	2.5 \pm 0.3	1.7 \pm 0.2
1-W66	0.027	<0.06	<0.02	3.3 \pm 0.3	2.3 \pm 0.2
1-W67	0.009	<0.06	6.4 \pm 0.64	2.4 \pm 0.3	1.6 \pm 0.2
1-SS68	159.0	0.61 \pm 0.06	0.60 \pm 0.06	1.2 \pm 0.2	0.8 \pm 0.1
1-W69	0.008	1.5 \pm 0.15	<0.02	6.8 \pm 0.7	4.6 \pm 0.5
1-SS70	165.5	0.46 \pm 0.07	0.82 \pm 0.08	1.8 \pm 0.2	1.2 \pm 0.1
1-W71	0.011	<0.06	<0.02	2.7 \pm 0.3	1.8 \pm 0.2
1-SS72	19.6	<0.06	2.46 \pm 0.25	2.2 \pm 0.2	1.5 \pm 0.2
1-W73	0.008	<0.06	<0.02	2.5 \pm 0.3	1.7 \pm 0.2
1-SS74	121.6	0.78 \pm 0.08	0.63 \pm 0.06	1.0 \pm 0.1	0.7 \pm 0.1
1-W86	12.3	<0.06	1.06 \pm 0.11	135.0 \pm 13.5	54.7 \pm 5.5 ^c
1-W90	109.1	0.65 \pm 0.06	0.85 \pm 0.08	4.4 \pm 0.5	3.0 \pm 0.3
1-W91	0.29	-	<0.02	10.3 \pm 1.0	4.2 \pm 0.4 ^c
1-W92	10.2	2.03 \pm 0.20	1.26 \pm 0.13	2.8 \pm 0.3	1.9 \pm 0.2
1-W93	13.0	<0.06	1.80 \pm 0.18	5.1 \pm 0.5	3.5 \pm 0.4
1-W95	5.1	<0.06	7.11 \pm 0.71	5.1 \pm 0.5	3.5 \pm 0.4

^a Conversion for Appendix 5.^b BDL = Below Detectable Limits^c Depleted uranium concentration factor of 0.405 as in Table 4.

TABLE 6

URANIUM MASS SPECTROMETRIC ANALYSES
OF SELECTED SOIL SAMPLES

Sample No.	U Isotopic, Atom %				
	233	234	235	236	238
1-S47-A	<0.001	0.001	0.228	0.005	99.766
D	<0.001	0.001	0.235	0.005	99.759
1-S48-A	<0.001	0.001	0.237	0.005	99.757
D	<0.001	0.001	0.273	0.005	99.271
1-S49-A	<0.001	0.001	0.230	0.005	99.764
D	<0.001	0.001	0.236	0.005	99.758
1-S50-A	<0.001	0.001	0.227	0.005	99.767
D	<0.001	0.001	0.228	0.005	99.766
1-S76	<0.001	0.001	0.226	0.005	99.768
1-S103-A	<0.001	0.001	0.226	0.005	99.768
1-S105-A	<0.001	0.001	0.225	0.006	99.768
1-R96	<0.001	0.005	0.711	<0.001	99.283

TABLE 7

ESTIMATED VOLUME, MASS AND RADIOACTIVITY OF
MATERIAL GENERATED BY REMEDIAL ACTION^a

Variable	Average (Option #1)	Maximum (Option #2)
Area (m ²)	6	6
Depth (m)	1	2
Volume (m ³)	6	12
Volume (ft ³)	212	424
Density (g/cm ³)	1.5	1.5
Mass (kg)	9,000	18,000
Mass (lbs)	20,000	40,000
Radioactivity (pCi/g)	240	26,000
Total Curies	0.0022	0.47

^a Assumptions on which calculations are made are specified in the text and in Appendix 7.

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 cm² (AC-21) or 325 cm² (AC-22) in area, with a thin double-aluminized Mylar window (~ 0.85 mg/cm²).

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 the beta mode, the detector responds to alpha and beta particles 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. This instrument is calibrated in the alpha mode with a flat-plate infinitely-thin NBS traceable ²³⁹Pu standard, and in the beta mode with a flat-plate infinitely-thin NBS traceable ⁹⁰Sr-⁹⁰Y standard. The PAC-4G-3 instruments are calibrated to an apparent 50% detection efficiency. An input discriminator threshold sensitivity of 1.5 mV is used with these detectors.

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, thus making the window density ~7 mg/cm². At this density, the instrument is not sensitive to alpha particles. 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. Another reading is obtained with the probe held 1 m from the contaminated area. This instrument is calibrated in mR/h with a ²²⁶Ra standard 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 ²³⁹Pu (17 keV), ²⁴¹Am (59.5 keV) and ²³⁵U (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 for measuring fields of low-level radiation between 10 $\mu\text{R/h}$ and 5000 $\mu\text{R/h}$. This instrument is also calibrated with a ^{226}Ra standard 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. This instrument 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 3-h 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 for counting 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 (PC-5 or PC-3A) gas-flow proportional counter. These instruments have been modified to contain a double-aluminized Mylar spun top. 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. These instruments are calibrated by determining the input sensitivity using an alpha source.

III. AIR SAMPLING DEVICE

Air samples were collected using a commercially available (ANL-modified filter queen) vacuum cleaner identified as a "Princess Model." The air was drawn through a filter media at a flow rate of 40 m^3/h . The filter media consist of 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-micron 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 (0.5 to 0.8 micron) filter paper

APPENDIX 1
(cont'd.)

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 α 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 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 cm² (dis/min-100 cm²) and the derivation of those factors are listed below.

A. Conversion Factors

	<u>PAC-4G-3</u>		<u>Floor Monitor (FM-4G)</u>	
	<u>Alpha</u>	<u>Beta</u>	<u>Alpha</u>	<u>Beta</u>
To 100 cm ²	1.96	1.96	0.31	0.31
cts/min per dis/min for ⁹⁰ Sr- ⁹⁰ Y	-	2	-	2
cts/min per dis/min for ²³⁹ Pu	2	-	2	-
cts/min per dis/min for normal U	3.1	3.0	2.9	2.9
cts/min per dis/min ²²⁶ Ra plus daughters	1.6	1.7	1.6	1.7

B. Derivation of Conversion Factors. Floor Monitor

Window Area: ~ 325 cm²

Conversion to 100 cm² = 0.31 times Floor Monitor readings

. PAC-4G-3

Window Area: ~ 51 cm²

Conversion to 100 cm² = 1.96 times PAC reading

. 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/cm²)} 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.

APPENDIX 2
(cont'd.)

The PAC-4G-3 and PC counters are calibrated (see Appendix 1) using a flat-plate, infinitely thin ^{226}Ra (including all 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 1.9×10^4 cts/min, or $1.9 \times 10^4 \div 0.51^* = 3.8 \times 10^4$ 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 2.3×10^4 cts/min at contact. The conversion factor for cts/min to dis/min for the PAC instrument is $3.8 \times 10^4 \div 2.3 \times 10^4 = 1.6$ dis/min alpha to 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 contrast with the chamber, the count was found to be 1.25×10^4 cts/min or $1.25 \times 10^4 \div 0.50 = 2.45 \times 10^4$ 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 1.45×10^4 cts/min. This indicates a conversion factor of $2.45 \times 10^4 \div 1.45 \times 10^4 = 1.7$ dis/min alpha to 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 mode.

A similar method was used to determine the conversion factors for normal uranium.

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)

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

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.

*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 $g \times \text{bf} \times \text{sa} \times \text{waf}$ is 0.51.

APPENDIX 2
(Cont'd.)

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)

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

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

I. RADON CONCENTRATIONS

The following postulates are 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) of 1.0 is used for the alpha activity.
- 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 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 are 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) are all 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 minute) based on the ThB half-life.
- Q. For all practical purposes 36% of the ThC (alpha branch) and the ThC' decay at the decay rate of ThB which is 638.4 minute.
- R. The counter does not differentiate between the ThC alphas and the ThC' alphas.

The following postulates are 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 are all 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 decay rate of 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 is 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 (minute) from end of sampling period to counting interval (usually \approx 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-m³)

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

f = Sampling rate (m³/min = m³/h x 1 h/60 minutes)

t_s = Length of sampling time (minute)

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

$t_{\frac{1}{2}}$ = Half-life of isotope or controlling parent (minute)

III. ACTINON CORRECTION

Since the actinon (²¹⁹Rn) progeny (AcA, AcB & AcC) decays at the AcB half-life of 36 minutes it cannot be distinguished from the radon (²²²Rn) 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 (0.5 to 0.8 micron) filter media. The sample rate is approximately

APPENDIX 3
(cont'd.)

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 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 fraction 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 = 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

APPENDIX 3
(cont'd.)

For long-lived activity:

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

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

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:

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

$$A_o = \frac{1816}{\exp - \frac{0.693 \times 100}{36}} = 12,454 \text{ dis/min}$$

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

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

APPENDIX 3
(cont'd.)

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.

$$B_1 = 44$$

$$B_1/0.84 = 52 \text{ counts}$$

$$B_2 = 601 \text{ counts}$$

$$\text{Actinon} = 52/653 = 0.08$$

$$\text{Radon} = 601/653 = 0.92$$

$$C(\text{Rn}) = C(36) \times \text{Radon}\% = 669.7 \times 0.92 = 616.1 \text{ dis/min-m}^3$$

$$C = C(36) \times \text{Actinon}\% = 669.7 \times 0.08 = 53.6 \text{ dis/min-m}^3$$

Since we assume that on the average half of the progeny is not adhered to the airborne particulates, the above concentrations are then multiplied by 2 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

$$C(\text{L}) = 0.19 \text{ dis/min-m}^3$$

$$C(\text{Tn}) = 7.6 \text{ dis/min-m}^3 \times 2 = 15.2 \text{ dis/min-m}^3$$

$$C(\text{An}) = 53.6 \text{ dis/min-m}^3 \times 2 = 107.2 \text{ dis/min-m}^3$$

$$C(\text{Rn}) = 616 \text{ dis/min-m}^3 \times 2 = 1232 \text{ dis/min-m}^3$$

These would then be the resulting concentrations in dis/min-m³. To convert to pCi/l, divide the concentrations by 2.2 x 10³.

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

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

APPENDIX 3
(cont'd.)

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

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

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 weighed. Each soil 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 (approximately 48 hours). Visual inspection of the soil is sufficient to determine when the soil is dry. 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 (1½" x 1½" 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 micron) 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) and radiochemical analyses for uranium. A 5-g sample of the sieved material is transferred to a 1-oz Nalgene bottle. This sample is used for the determination of uranium by laser fluorometry. 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 then rinsed with tap water (three times) followed by demineralized water (three times) 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 using warm air until absolutely dry. Room air is blown through the mill hars with a hose attached to a fume hood.

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 bottles 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 furnace. The first trap is immersed in an ordinary ice bath (0°C); the second trap is immersed in a CO₂-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 multi-channel analyzer. This analysis allows for a quantitative determination of the ²²⁶Ra decay chain (via the 609 keV γ -ray of ²¹⁴Pb) and the ²³²Th decay chain (via the 908 keV γ -ray of ²²⁸Ac as well as any other gamma emitting radionuclide (e.g. ¹³⁷Cs) 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 ²³⁴Th spike for yield determination) followed by plating a thin source of the radiochemically separated thorium and determining the thorium isotopes (²²⁸Th and ²³²Th) 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 it is surmised that depleted or enriched uranium might be present.

APPENDIX 5

CALCULATION OF 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 specific activities and corresponding half-lives for ^{235}U and ^{238}U were taken from A. H. Jaffey, et al. Phys. Rev. C 4 1889 (1971). The corresponding data for ^{234}U was taken from "Table of Isotopes" 7th Edition by C. M. Lederer and V. S. Shirley (1978). The percent abundances were taken from N. E. Holden, BNL-NCS-50605 (1977). Atomic weights were taken from the Handbook of Chemistry and Physics, 52nd Edition (1971).

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	0.7200	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 $(\text{wt } \%)_i =$

$$\frac{(\text{atom } \%)_i (\text{atomic weight})_i}{\sum_j (\text{atom } \%)_j (\text{atomic weight})_j} = \frac{(\text{atom } \%)_i (\text{atomic weight})_i}{238.02985}$$

Specific activity for normal uranium:

$$\begin{aligned} 0.746 \times 0.99284 \times 2 &= 1.481 \text{ dis/min-}\mu\text{g from } ^{234}\text{U} \text{ \& } ^{238}\text{U} \\ 4.798 \times 0.00711 &= 0.0341 \text{ dis/min-}\mu\text{g from } ^{235}\text{U} \\ &1.515 \text{ dis/min-}\mu\text{g for normal U} \end{aligned}$$

or $(1.515 \text{ dis/min-}\mu\text{g}) / (2.22 \text{ dis/min-pCi}) = 0.683 \text{ pCi}/\mu\text{g}$ where ^{234}U is assumed to be in secular equilibrium with the ^{238}U parent. In this case 2.25% of the total activity is due to ^{235}U and 48.87% each is due to ^{234}U and ^{238}U .

Calculation of the specific activity of other than normal mixtures of uranium isotopes are performed in a similar manner. For example, neutron activated reprocessed uranium having the isotopic composition 238(0.99766), 236(0.00005), 235(0.00228), 234(0.00001) as determined by mass spectrometry would have the following specific activity:

$$\begin{aligned} 0.99766 \times 0.746 &= 0.7443 \text{ dis/min-}\mu\text{g from } ^{238}\text{U} \\ 0.00005 \times 1.436 \times 10^2 &= 0.0072 \text{ dis/min-}\mu\text{g from } ^{236}\text{U}^a \\ 0.00228 \times 4.798 &= 0.0109 \text{ dis/min } \mu\text{g from } ^{235}\text{U} \\ 0.00001 \times 1.387 \times 10^4 &= 0.1387 \text{ dis/min } \mu\text{g from } ^{234}\text{U} \\ &0.9011 \text{ dis/min } \mu\text{g U total} \end{aligned}$$

corresponding to: $(0.9011 \text{ dis/min } \mu\text{g}) / (2.22 \text{ dis/m pCi}) = 0.406 \text{ pCi}/\mu\text{g U}$

^aThe half-life for ^{236}U (2.342×10^7 yr) and the corresponding specific activity (1.436×10^2) dis/min- μg) were taken from K. F. Flynn, et al J. Inorg. Nucl. Chem. 34 1121 (1972).

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 non-occupational MPC (Note 2) is 2×10^{-13} Ci/m ³ or less or for which the nonoccupational MPC (Note 4) is 2×10^{-7} Ci/m ³ or less	227Ac 241, 242 ^m , 243Am 249, 250, 251, 252Cf 243, 244, 245, 246, 247, 248Cm 125, 129I 237Np 231Pa 210Pb 238, 239, 240, 242, 244Pu 226, 228Ra 228, 230Th	20	Nondetectable (Note 3)
2	Those nuclides not in Group 1 for which the nonoccupational MPC (Note 2) is 1×10^{-12} Ci/m ³ or less for which the nonoccupational MPC (Note 4) is 1×10^{-6} Ci/m ³ or less	254Es 256Fm 126, 131, 133I 210Po 223Ra 90Sr 232Th 232U	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 cm² 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 cm², if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i=1}^n S_i \geq L$, where S_i is the dis/min-100 cm² 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 cm² 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 cm².
- (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 nat 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 cm² 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 cm², if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i=1}^n S_i \geq L$, where S_i is the dis/min-100 cm² 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 cm² 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 cm².
- (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:
 - a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

APPENDIX 6
(Cont'd.)

- 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 (nu- clides with decay modes other than alpha emission or spontaneous fission) except ⁹⁰ Sr and others noted above.	5000 dis/min-100 cm ² βγ	15,000 dis/min-100 cm ² βγ	1000 dis/min-100 cm ² βγ

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(Cont'd.)

TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LEVELS

- ^aWhere surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.
- ^bAs 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.
- ^cMeasurements 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.
- ^dThe maximum contamination level applies to an area of not more than 100 cm².
- ^eThe amount of removable radioactive material per 100 cm² 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.
- ^fThe 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.

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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.

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(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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(Cont'd.)

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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(Cont'd.)

(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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(cont'd.)

- (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.

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.

APPENDIX 6
(cont'd.)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,

(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

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

**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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(cont'd.)

(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 derived 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.

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 or near the surface of the earth including, but not limited to silts, clays, sands, gravel, and small rocks.

APPENDIX 6
(cont'd.)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."

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

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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 effectiveness 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.

(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.

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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 Criteria 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 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 where 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

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(cont'd.)

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)

	Inhalation	Ingestion		External Radiation	All Pathways ^c
		Home Gardener	Full Diet		
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 ²⁴¹Am and ^{239,240}Pu are available from EPA recommendations, and a soil limit for ²²⁶Ra 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

Type of Exposure	Annual Dose Equivalent or Dose Commitment	
	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)

CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

Element (atomic number)	Isotope (I) soluble (S)	Table I Controlled Area		Table II Uncontrolled Area	
		Column 1 Air (pCi/l)	Column 2 Water (pCi/l)	Column 1 Air (pCi/l)	Column 2 Water (pCi/l)
Radon (86)	Rn 220 S	300		10	
	Rn 222 S	100		3	

APPENDIX 7

ESTIMATED EXTENT OF CONTAMINATION

Estimates of the extent of the contamination at the assessed site are based on the total volume, mass, and quantity of radioactive material in the contaminated area. The volume is the product of the surface area and the depth of the contamination. The mass is the product of the volume and the density of the material. A density of 1.5 g/cm^3 is used for soil. The concentration (pCi/g) of the specific radioisotope is determined by radiochemical analysis of the soil. The total quantity of radioactive material is the product of the concentration of the specific radioisotope and the total mass of material.

Often there is more than one contaminant in the soil (or contaminated material) and the contaminants are not uniformly distributed throughout the material. In these cases, it is necessary to estimate the fraction of the material containing each contaminant in order to assess the total quantity of the radioactive material. This estimate of the fraction of the material containing each contaminant is based on the radiochemical analysis of randomly selected samples.

Estimates of the extent of contamination are usually determined for averaged (Option 1) and maximum or worst-case (Option 2) conditions. A sample calculation for the extent of contamination in a typical site is as follows:

$$\begin{aligned} \text{Volume (Average)} &= 34,800 \text{ ft}^2 \text{ (area)} \times 3.6 \text{ ft (avg. depth)} = 125,000 \text{ ft}^3 \\ &= 3,550 \text{ m}^3 \end{aligned}$$

$$\begin{aligned} \text{Volume (Maximum)} &= 34,800 \text{ ft}^2 \text{ (area)} \times 9 \text{ ft (max. depth)} = 314,000 \text{ ft}^3 \\ &= 8,880 \text{ m}^3 \end{aligned}$$

$$\text{Mass (Average)} = 3,550 \text{ m}^3 \times 1,500 \text{ kg/m}^3 = 5.33 \times 10^6 \text{ kg}$$

$$\text{Mass (Maximum)} = 8,880 \text{ m}^3 \times 1,500 \text{ kg/m}^3 = 1.33 \times 10^7 \text{ kg}$$

Estimated Total Activity for ^{226}Ra (chain)

$$\text{Average: } 5.33 \times 10^6 \text{ kg} \times 14 \times 10^{-12} \text{ Ci/g} \times 10^3 \text{ g/kg} \times .05 \text{ (fraction)*} = 0.004 \text{ Ci}$$

$$\text{Maximum: } 1.33 \times 10^7 \text{ kg} \times 16 \times 10^{-12} \text{ Ci/g} \times 10^3 \text{ g/kg} \times .05 \text{ (fraction)*} = 0.011 \text{ Ci}$$

*This represents the estimate of the fraction of the total mass contaminated with the ^{226}Ra chain.

APPENDIX 8

EVALUATION OF RADIATION EXPOSURESINTRODUCTIONA. 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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(cont'd.)

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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(cont'd.)

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 gas and coal) and nuclear reactors; mining, milling, and tailings piles; television sets, smoke detectors, and watch dials could be responsible for an

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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 doses 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 his 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 led 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 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) reports. 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. The risk

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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 individual 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.

APPENDIX 8
(cont'd.)DOSE DETERMINATION CALCULATIONExternal Exposure

External penetrating radiation dose rates are measured on contact with an end-window beta-gamma Geiger-Mueller (GM) detector (7 mg/cm² window), and at 1 meter with a NaI crystal detector (1 in diameter by 1 in thick) manufactured by Eberline Corporation (PRM-7 μ R Meter). For the purpose of these calculations, the following conservative assumptions are made. First, it is assumed that the half-life of the contaminant is long and, therefore, the dose rate is constant with respect to time. Second, it is assumed that a person is stationary at the location of maximum dose for 40 hours per week for 50 weeks per year. For such a situation, the annual dose (A) for a 0.3 mR/h radiation field (about ten times normal background) would be:

$$A = 40 \text{ hr/week} \times 50 \text{ weeks/yr} \times 0.3 \text{ mR/h} = 600 \text{ mR/yr}$$

For the purposes of this example it is assumed that one milliRoentgen of penetrating radiation is equivalent to one millirem of dose. Hence, the maximum dose for this case would be 600 mrem. This value is then compared with the allowable limit of 500 mrem per year for a person non-occupationally exposed. ⁽¹⁾

Internal Exposure

The internal radiological hazard from inhalation/ingestion of contamination is assessed by postulating hypothetical "worst case" scenarios. To this end two cases are considered. The first case is based on the situation whereby a child would eat 100 g per year of the contaminated soil. The second case assumes a home gardener would rototill the contaminated soil (dry) to a 1-ft depth for a working day (eight hours) once a year. For this latter case, a resuspension factor of 10^{-6} m^{-1} , a breathing rate of 9.6 m³/working day and a soil density of 1.5 g/cm³, are used. In both cases it is assumed that the average concentration of contaminants in the soil is equal to the maximum measured value (a conservative assumption). All calculations are based on methods outlined in ORNL/NUREG/TM-190, Vol. 3. ⁽²⁾ These calculations approximate the ICRP-30 guidelines for hazard analysis.

The adult inhalation and ingestion dose commitment factors for the bone, lung and total body from ²³⁸U, ²³⁵U, ²³⁴U and ²³²Th (and all their significant daughters), as determined in Reference 2, are given in Appendix 8 Table 1. Normal uranium is assumed to be composed of 2.26% ²³⁵U, and 97.74% ²³⁴U and ²³⁸U (in equilibrium) by activity.

An example calculation, based on the above scenarios and assuming a soil contamination of 5 pCi/g of radium gives the following hazard levels (50-year dose commitment):

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Ingestion (consumption of 100 g/yr of soil):

$$\begin{aligned} \text{Bone:} & \quad 6.5 \times 10^{-2} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 100 \text{ g} = 32.5 \text{ mrem} \\ \text{Total Body:} & \quad 5.5 \times 10^{-3} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 100 \text{ g} = 2.75 \text{ mrem} \end{aligned}$$

Inhalation (rototilling and breathing ensuing aerosol):

$$\begin{aligned} \text{Lung:} & \quad 1.1 \times 10^{-1} \text{ mrem/pCi} \times 2.285 \text{ pCi/m}^3 \times 9.6 \text{ m}^3 = 2.4 \text{ mrem} \\ \text{Bone:} & \quad 9.2 \times 10^{-2} \text{ mrem/pCi} \times 2.285 \text{ pCi/m}^3 \times 9.6 \text{ m}^3 = 2.0 \text{ mrem} \\ \text{Total Body:} & \quad 9.5 \times 10^{-3} \text{ mrem/pCi} \times 2.285 \text{ pCi/m}^3 \times 9.6 \text{ m}^3 = 0.2 \text{ mrem} \end{aligned}$$

These values are then compared with the allowable limit of 170 mrem per year for a person non-occupationally exposed. ⁽¹⁾

TABLE 1

FIFTY-YEAR DOSE COMMITMENT FACTORS (Inhalation/Ingestion)^a
(mrem/pCi assimilated)

Nuclide ^b	Inhalation			Ingestion	
	Bone	Lung	Total Body	Bone	Total Body
²³⁸ U	7.1×10^{-3}	4.8×10^{-1}	1.5×10^{-2}	2.8×10^{-4}	2.1×10^{-5}
²³⁴ U	7.9×10^{-3}	5.4×10^{-1}	1.6×10^{-2}	3.1×10^{-4}	2.4×10^{-5}
²³⁰ Th	3.1×10^{-1}	5.3×10^{-1}	3.8×10^{-2}	1.2×10^{-3}	9.2×10^{-5}
²²⁶ Ra	4.9×10^{-2}	5.6×10^{-2}	4.7×10^{-3}	4.3×10^{-2}	3.4×10^{-3}
²¹⁰ Po	8.1×10^{-4}	4.6×10^{-2}	1.3×10^{-3}	5.2×10^{-4}	4.1×10^{-4}
²¹⁰ Pb	4.2×10^{-2}	6.2×10^{-3}	3.5×10^{-3}	2.1×10^{-2}	1.7×10^{-3}
Total Chain	4.2×10^{-1}	1.7	7.9×10^{-2}	6.6×10^{-2}	5.6×10^{-3}
²²⁶ Ra Chain	9.2×10^{-2}	1.1×10^{-1}	9.5×10^{-3}	6.5×10^{-2}	5.5×10^{-3}
²³⁸ U+ ²³⁴ U	1.5×10^{-2}	1.0	3.1×10^{-2}	5.9×10^{-4}	4.5×10^{-5}
²³⁵ U	7.2×10^{-3}	4.8×10^{-1}	1.5×10^{-2}	2.8×10^{-4}	2.2×10^{-5}
²³¹ Pa	9.6×10^{-1}	5.9×10^{-1}	1.4×10^{-1}	1.8×10^{-2}	2.1×10^{-3}
²²⁷ Ac	5.4×10^{-1}	1.0	1.0×10^{-1}	1.2×10^{-2}	1.3×10^{-3}
²²³ Ra	9.6×10^{-4}	4.6×10^{-2}	8.6×10^{-4}	1.2×10^{-3}	2.3×10^{-4}
²²⁷ Th	4.8×10^{-4}	6.9×10^{-2}	1.1×10^{-3}	2.7×10^{-5}	5.1×10^{-6}
Total Chain	1.5	2.2	2.6×10^{-1}	3.2×10^{-2}	3.7×10^{-3}
²³² Th	3.3×10^{-1}	4.5×10^{-1}	3.8×10^{-2}	1.3×10^{-3}	9.6×10^{-5}
²²⁸ Th	4.4×10^{-2}	7.2×10^{-1}	1.9×10^{-2}	4.5×10^{-4}	3.8×10^{-5}
²²⁸ Ra	2.9×10^{-2}	4.8×10^{-3}	2.5×10^{-3}	2.1×10^{-2}	1.7×10^{-3}
²²⁴ Ra	3.0×10^{-4}	8.8×10^{-3}	1.8×10^{-4}	4.0×10^{-4}	7.5×10^{-5}
²¹² Pb	2.6×10^{-5}	1.8×10^{-3}	2.9×10^{-5}	1.8×10^{-5}	2.6×10^{-6}
Total Chain	4.0×10^{-1}	1.2	6.0×10^{-2}	2.3×10^{-2}	1.9×10^{-3}

^aData taken from Reference 2.

^bNuclides in the chain that contribute negligibly (e.g. $<10^{-6}$ mrem) have not been included.

APPENDIX 8
(cont'd.)REFERENCES

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