MA.02-9

DOE/EV-0005/37

ANL-OHS/HP-83-106

FORMERLY UTILIZED MED/AEC SITES REMEDIAL ACTION PROGRAM

RADIOLOGICAL SURVEY OF THE FORMER WATERTOWN ARSENAL PROPERTY SITE 34 AND SITE 41 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

Argonne National Laboratory, with facilities in the states of Illinois and Idaho, is owned by the United States government, and operated by The University of Chicago under the provisions of a contract with the Department of Energy.

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: A05
Microfiche copy: A01

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

DOE/EV-0005/37
ANL-OHS/HP-83-106

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

FORMERLY UTILIZED MED/AEC SITES REMEDIAL ACTION PROGRAM

RADIOLOGICAL SURVEY OF THE FORMER WATERTOWN ARSENAL PROPERTY SITE 34 AND SITE 41 WATERTOWN, MASSACHUSETTS

Prepared by

R. A. Wynveen	Associate Division Director, OHS
W. H. Smith	Senior Health Physicist
C. M. Sholeen	Health Physicist
A. L. Justus	Health Physicist
K. F. Flynn	Health Physicist

Radiological Survey Group
Health Physics Section
Occupational Health and Safety Division

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 that were 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 determine current radiological conditions. The results of a radiological survey of Building Site 421 were contained in a final report (DOE/EV-0005/19) issued in February 1980. The results of surveys of Building Site 34 and Building Site 41 are presented in this report. The results of a radiological survey of the area north of Arsenal Street identified as the "GSA" property is presented in a companion report (ANL-OHS/HP-83-108).

During the MED/AEC era, work involving radioactive materials was conducted at various sites within the arsenal complex. Building 34 housed a uranium machine shop, and a portion of Building 41 contained a foundry that was used for uranium work. Information provided by site personnel indicated that only depleted uranium was used in these buildings. Results of radiological analyses of contaminated material found at these sites indicated depleted uranium with uranium-236. Both buildings have been razed. The remnants still in place consist of the concrete floor slabs, access drives, and underground utility service trenches. This area is currently under the control of the Watertown Redevelopment Authority of Watertown, Massachusetts.

During the period from June 25 through July 1, 1981, the Argonne National Laboratory (ANL), Occupational Health and Safety Division (OHS) Radiological Survey Group (RSG), conducted a radiological survey of Building Sites 34 and 41

at the direction of the U.S. Department of Energy. Significant levels of contamination were found at 33 locations on the pad of Site 34 and in 5 out of 15 soil corings from the perimeter of the pad. No contamination was found on the pad of Site 41, however, two-thirds of this pad was covered with soil up to 4 ft thick. One of the 14 soil corings taken adjacent to the pad of Site 41 had elevated levels of uranium.

Levels of contamination in excess of criteria, as identified in ANSI 13.12 and NRC Guidelines, were found at this site. The analyses of the samples from the sewer access points also revealed uranium and radium-226 anomalies. Therefore, according to NRC guidelines dated July 1982, it must be concluded that they are contaminated.

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*, J. D. Thereon, R. Rodriguez, and D. W. Reilly.

*Now at Los Alamos Scientific Laboratory.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

FORMERLY UTILIZED MED/AEC SITES REMEDIAL ACTION PROGRAM
RADIOLOGICAL SURVEY OF THE FORMER WATERTOWN ARSENAL PROPERTY
SITE 34 AND SITE 41
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). The Watertown Arsenal ceased operations in June 1967, and responsibility for release of areas in the former Watertown Arsenal properties was transferred, along with the AEC Source Materials License SUB-238, to the Army Materials and Mechanics Research Center (at that time called the Army Materials Research Agency). Subsequently, control of this area was ceded to the Watertown Redevelopment Authority, Watertown, Massachusetts.

As a result of the MED/AEC activities, portions of the property had become radiologically contaminated. Subsequent decontamination was carried out 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 of these sites is insufficient to establish conformity with present guidelines. Hence, the Argonne National Laboratory (ANL), Occupational Health and Safety Division (OHS) Radiological Survey Group (RSG), was requested by the U.S. Department of Energy (DOE) to perform a radiological assessment of these properties to ascertain the extent of any remaining contamination.

The property included in the survey reported in this document consisted of the former sites of Buildings 34 and 41, both located south of Arsenal Street (see Fig. 2). During the MED/AEC era, Building 34 housed a uranium machine shop, and a portion of Building 41 contained a foundry that was used for uranium work. According to information provided by site personnel only depleted uranium was used in these buildings. Subsequent mass-spectrometric analyses indicated depleted uranium with uranium-236. Both buildings have been razed. The remnants still in place consist of the concrete floor slabs, access drives, and underground utility service trenches.

The ANL/OHS-RSG performed this assessment during the period June 25 through July 1, 1981. At the time of the survey, the pad at Site 34 was rented to an automobile distributor for use as a parking lot. The pad at Site 41 was in the process of being covered with 4 ft of fill material in anticipation of its use as a parking facility. This entire area is gradually being converted to sites for rental living units and commercial business use.

The site of Building 421 (see Fig. 2) underwent a radiological assessment by the ANL/OHS-RSG in 1979. The results were published in a DOE report (DOE/EV-0005/19). An area north of Arsenal Street (see Fig. 2), presently in use by a variety of Federal agencies including the General Services Administration (GSA), has also undergone a radiological assessment by the ANL/OHS-RSG. The results of that assessment are presented in a companion report (ANL-OHS/HP-83-108).

SURVEY AND ANALYTICAL TECHNIQUES

General

The methods used in this radiological survey included making instrument surveys to measure radiation levels of all accessible surface areas; collection and analyzing environmental soil samples at representative locations, as well as sewer, water, and sludge samples where appropriate; and measuring ambient radiation levels at the 3-ft level utilizing a pressurized ion chamber.

The overall dimensions of the building pad at Site 34 are 497 ft long and 80 ft wide. An irregular-shaped loading dock area is located at the east end extending 15 ft to 40 ft. For survey and identification purposes, the concrete pad and the surrounding land was sectioned (mapped) into a network of 40-ft by 40-ft squares (see Fig. 3). The area surveyed included the concrete building pad, the loading dock area, and the surrounding soil surface out to about 20 ft from the edge of the concrete. Environmental soil samples (soil corings) were taken from selected locations surrounding the pad and sludge and water samples were taken from a sewer adjacent to the pad.

The pad remaining from Building 41 is irregular in shape with the approximate dimensions of 700 ft by 250 ft (see Fig. 4). This area also was sectioned (mapped) for identification and survey purposes into a network of 100-ft by 100-ft squares.

At the time of this survey, about two-thirds of the pad had been covered with fill material (see Fig. 4). A portion of the filled area had been leveled to a thickness of 4 ft. The remainder of the fill was in piles on the pad. The eastern third of the pad which was clear of fill was completely surveyed with all survey instruments. The remaining area, which was covered with fill, was also scanned in its entirety with the survey meters. Environmental soil samples (soil corings) were taken from selected locations surrounding the pad. Samples of sludge and water were taken from a sump connected to an adjacent sewer.

Instrumentation

Four types of portable survey instruments were used to conduct the direct radiological surveys. Propane gas-flow proportional detectors with window areas of 51 cm^2 and 325 cm^2 (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 μR meter) and calibrated with a ^{226}Ra standard source, were used to measure the ambient external penetrating radiation field in units of μR/h. An end-window Geiger-Mueller (GM) detector (Eberline HP-190 with a 7 mg/cm^2 window and Eberline 530 electronics), calibrated with a ^{226}Ra standard source, was used to measure the contact exposure rate (mR/h) of 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 or HPGe detector coupled to a multichannel analyzer. This instrumentation is also described in Appendix 1.

Soil Corings

Environmental soil samples (4-in-diameter by 12-in-deep corings) were taken from selected locations. The samples were collected using a 4-in-diameter, 6-in-long, right-circular-cylinder cutting tool, commonly used to cut holes on golf greens. Each soil core was taken in four segments. 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 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 analysis (see Fig. 5).

ANALYSIS OF SURVEY RESULTS

General

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 cm^2. After subtraction of background, readings in the alpha mode also were converted to dis/min-100 cm^2. Two survey instruments were used. The first, identified as the floor monitor (FM), had a 325-cm^2 probe. The second, identified as a hand probe (HP), had a 51-cm^2 probe.

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 μR meter exposure-rate measurements include the instrument backgrounds of 0.03 mR/h and 7-10 μR/h, respectively.

Instrument Surveys

Surface radioactive contamination was identified at 33 locations in 7 grid blocks at Site 34. These results are detailed in Table 1 and the locations are identified in Figure 3. No specific spots of radioactive contamination were found on Site 41; however, the soil cover on most of the pad would certainly tend to mask any radioactive contamination extant on the pad.

Each location where radioactive contamination was detected was delineated as to the total area involved (see Table 1). At 21 locations, the contamination was localized to a spot of no more than 100 cm^2. In 12 other locations, the area of contamination varied between 900 cm^2 and 35,000 cm^2.

The entire areas of the locations identified as Site 34 and Site 41 were scanned at heights of about 1 m above the surface with a μR gamma survey instrument (PRM-7). No instrument readings above background (i.e., 7-10 μR/h) were observed. Integrated measurements of the ambient penetrating radiation field were also taken with a pressurized ioniziation chamber (Reuter-Stokes RSS-111) at two locations on Site 34 (see Fig. 6) and at three locations on Site 41 (see Fig. 4). These results are summarized in Table 2. No anomalous ambient radiation levels were detected.

Soil Corings

Environmental soil samples (soil corings) were obtained at 14 locations around Site 34 (see Fig. 6). The locations were chosen so as to encompass the perimeter of the concrete pad at site 34. These corings were identified as 1-S10 through 1-S23. An additional sample (1-S24) was taken from a filled-in floor drain trap located 100 ft east of Site 34 (see Fig. 6). Samples of contaminated chips and debris identified as 1-S9, 1-S42, and 1-S43 were taken from the surface of the pad (see Fig. 6). Gamma spectral analyses of these samples indicated normal uranium contamination similar to material currently in use at the site.

Soil corings were also obtained at 14 locations encompassing the perimeter of the pad at Site 41 (see Fig. 7). These corings were identified as 1-S26 through 1-S39. Considerable difficulty was experienced in obtaining these samples. A large fraction of the "soil" in this area consisted of rocks and debris. Hence, segregation between the fractions for each coring was difficult, if not impossible, to maintain. Consequently, any variations in the results from these fractions would be difficult to interpret. At two locations (1-S36 and 1-S38), it was not possible to sample the full 1-ft depth due to the presence of buried concrete.

Background soil corings were taken from two locations some distance from the Watertown Arsenal Site (see Fig. 8). Two corings (1-SB107 and 1-SB108) were taken from the Newton area, and two (1-SB109 and 1-SB110) were taken from the Stoneham area. These samples were used to determine background levels of radionuclides in soil for the Watertown region.

All soil samples were prepared as outlined in Appendix 4. Results of the uranium-fluorometric analyses and the gamma-spectral analyses for the soil samples are given in Table 3. No elevated levels of radionuclides in the ^{232}Th decay chain or the ^{226}Ra decay chain were found. However, five of the soil corings at Site 34 (1-S10, 1-S11, 1-S21, 1-S23, and 1-S24) and one of the soil corings at Site 41 (1-S28) had elevated levels of uranium. The presence of elevated uranium levels and background levels of the ^{226}Ra decay chain indicates that this material is uranium other than natural (as opposed to natural uranium, which is in equilibrium with its daughters). Mass spectrometric analyses revealed that the contaminant is depleted uranium with uranium-236 (see Table 5).

Water and Sludge Samples

Attempts were made to sample the water and sludge from sewers in the area. Various manhole covers located on the pad at Site 34 proved to be access to utility service trenches. A sewer located about 250 ft east of the pad was the closest drain to this site. A water sample (1-W25) and a sludge sample (1-SS25) were taken from this drain. What appeared to be a floor drain and sump leading to a sewer was located in grid block B-6 of Site 41. A water sample (1-W40) and a sludge sample (1-SS41) were taken from this drain. These water and sludge

samples were analyzed by gamma spectrometry and uranium fluorometry (see Appendix 4). The results of these analyses are given in Table 4. The ^{232}Th decay chain and ^{226}Ra decay chain activity levels were within the normal background range except for samples taken from the Site 41 drain. These had somewhat elevated levels of uranium (5.8 pCi/g) and ^{226}Ra (5.5 pCi/g) in the suspended solids from the sludge sample. In addition, an elevated level of uranium (12.0 pCi/g) was found in the suspended solids from the water sample.

ESTIMATED EXTENT OF CONTAMINATION

Modest levels of surface contamination were found on Site 34 (see Table 1). No surface contamination was found on Site 41; however, most of the surface was covered with 4 ft of earth so any contamination would be masked significantly.

Corings were taken from the perimeters of both sites. The highest levels of contamination measured in these corings were 16 pCi/g of depleted uranium at Site 34 and 8.7 pCi/g of depleted uranium at Site 41. One third of the corings at Site 34 (i.e., 5 of 15) were contaminated; whereas only 1 of 14 corings taken at Site 41 was contaminated. The highest level of contamination found (16 pCi/g) was about 8 times normal background. The details of the conversion of the analytical data (μg/g) to pCi/g for material containing varying mixtures of depleted uranium with uranium-236 and natural uranium is given in footnote "c" of Table 3.

For this evaluation, the following assumptions have been made:

1. There is no reason to suspect contamination beneath the concrete pads since these were in place prior to the initiation of work involving radioactivity.

2. About 13 m^2 of the concrete pad at site 34 are contaminated to an assumed average depth of 0.2 m (i.e. about 2.6 m^3 of concrete).

3. About one third (5 of 15 soil corings) of the perimeter of site 34 (10 m x 350 m) is contaminated to an estimated depth of 0.3 m (i.e. about 1200 m^3 of soil).

4. The highest level of contamination was 16 pCi/g of depleted uranium. Hence, this concentration is representative of a worst case scenario.

Based on these assumptions, about 2.6 m^3 (2×10^4 kg) of contaminated concrete and about 1200 m^3 (1.8×10^6 kg) of contaminated soil containing about 0.03 Ci of depleted uranium would be excavated and disposed of.

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 to the maximum radiation level.

The maximum radiation level observed was 7 mR/h on contact at location I-3 on Site 34. The annual radiation dose from this source would be:

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

This dose is based on the assumption that a person would stand on one spot for 40 hours per week, for 50 working weeks, an obviously improbable situation. The dose received under this assumption is well above the DOE 5480.1[1] limit of 500 millirems per year (whole body) for a person non-occupationally exposed (see Appendices 6 and 8). However, the actual likelihood of an individual's receiving a dose in excess of 500 millimrems per year from this source is extremely remote.

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 cases were considered. The first case was based on a situation in which a child would eat 100 g per year of contaminated soil. For the second case, a person was assumed to 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 and a breathing rate of 9.6 m^3/working day[(2)] were used. 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 16 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: mrem/pCi x pCi/g x g = mrem)

   normal uranium: Bone: (0.0003)(16)(100) = 0.5 mrem
   Total Body: $(2 \times 10^{-5})(16)(100) = 0.03$ mrem

2. Adult inhalation of aerosol:
   (units: mrem/pCi x pCi/m^3 x m^3 = mrem)

   normal uranium: Bone: (0.0075)(7.3)(9.6) = 0.5 mrem
   Total Body: (0.015)(7.3)(9.6) = 1.1 mrems
   Lung: (0.43)(7.3)(9.6) = 30 mrem

These levels do not pose a significant radiological hazard even with the "conservative" assumptions used in these scenarios (see Appendices 6 and 8).

CONCLUSIONS

Surface Contamination

Levels of contamination exceeding criteria as defined in ANSI 13.12 and the NRC Guidelines (see Appendix 6) were found. It should also be noted that contamination in concrete is difficult to quantify since depth concentration varies depending on the mode of application of the contaminant and the effects of weathering.

Soil

There is no standard that pertains directly to soil contamination for this specific site since the contaminant was found to be depleted uranium containing uranium-236. The maximum level of contamination found was 16 pCi/g. This

compares to values of 75 pCi/g for normal uranium (i.e. uranium exclusive of its natural daughters) and 15 pCi/g for ^{226}Ra (i.e. the natural radium daughter of ^{238}U) which have been submitted as proposed criteria.[5] In any case, criteria that are consistent with "As Low As Reasonably Achievable" (ALARA) concepts must be considered in evaluating contamination.

Sewer Systems

The access points to the sewer systems indicated radium-226 and/or uranium anomalies for Site 41. Therefore, in accord with the NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product Source or Special Nuclear Material" dated July 1982, the sewers are presumed to be contaminated.

REFERENCES

1. U.S. Department of Energy. 1981. "Requirements for Radiation Protection." DOE 5480.1, Chapter XI.

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).

5. "Radiological Guidelines for Application to DOE's Formerly Utilized Sites Remedial Action Program." ORO-831, March 1983.

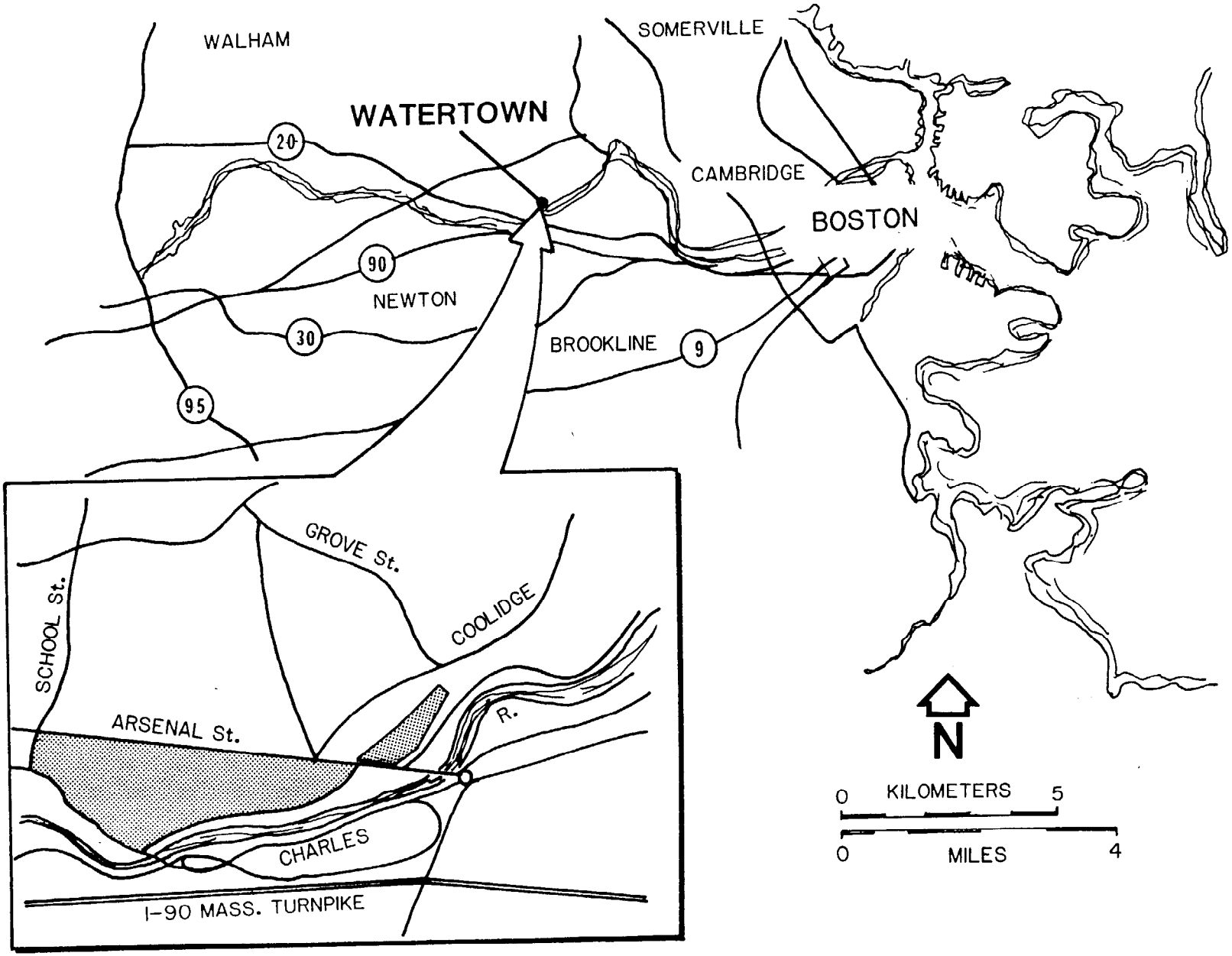

Fig. 1. Watertown Arsenal Site Area.
ANL-HP Dwg. No. 82-10

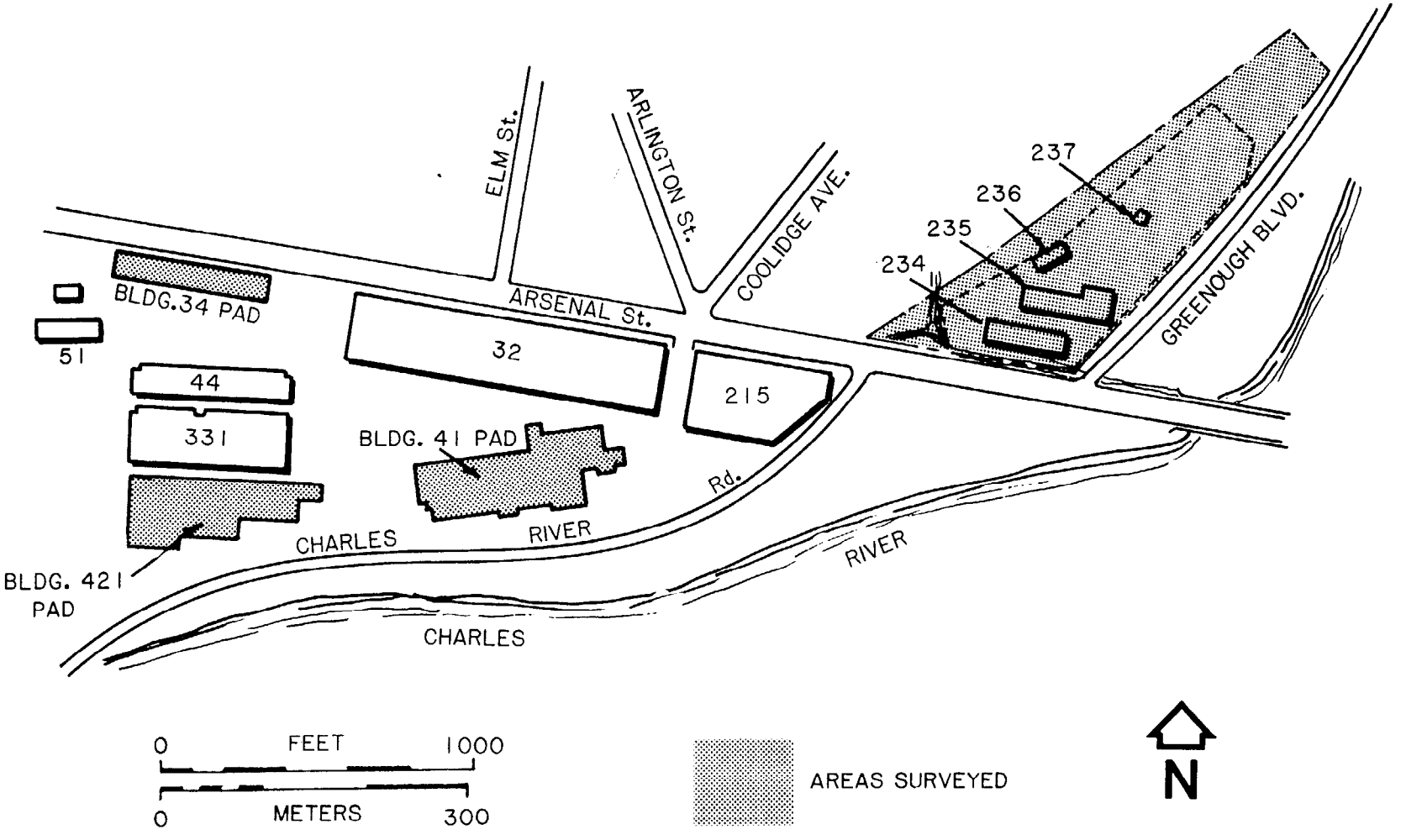

Fig. 2. Watertown Arsenal - Survey Site.
ANL-HP Dwg. No. 82-6

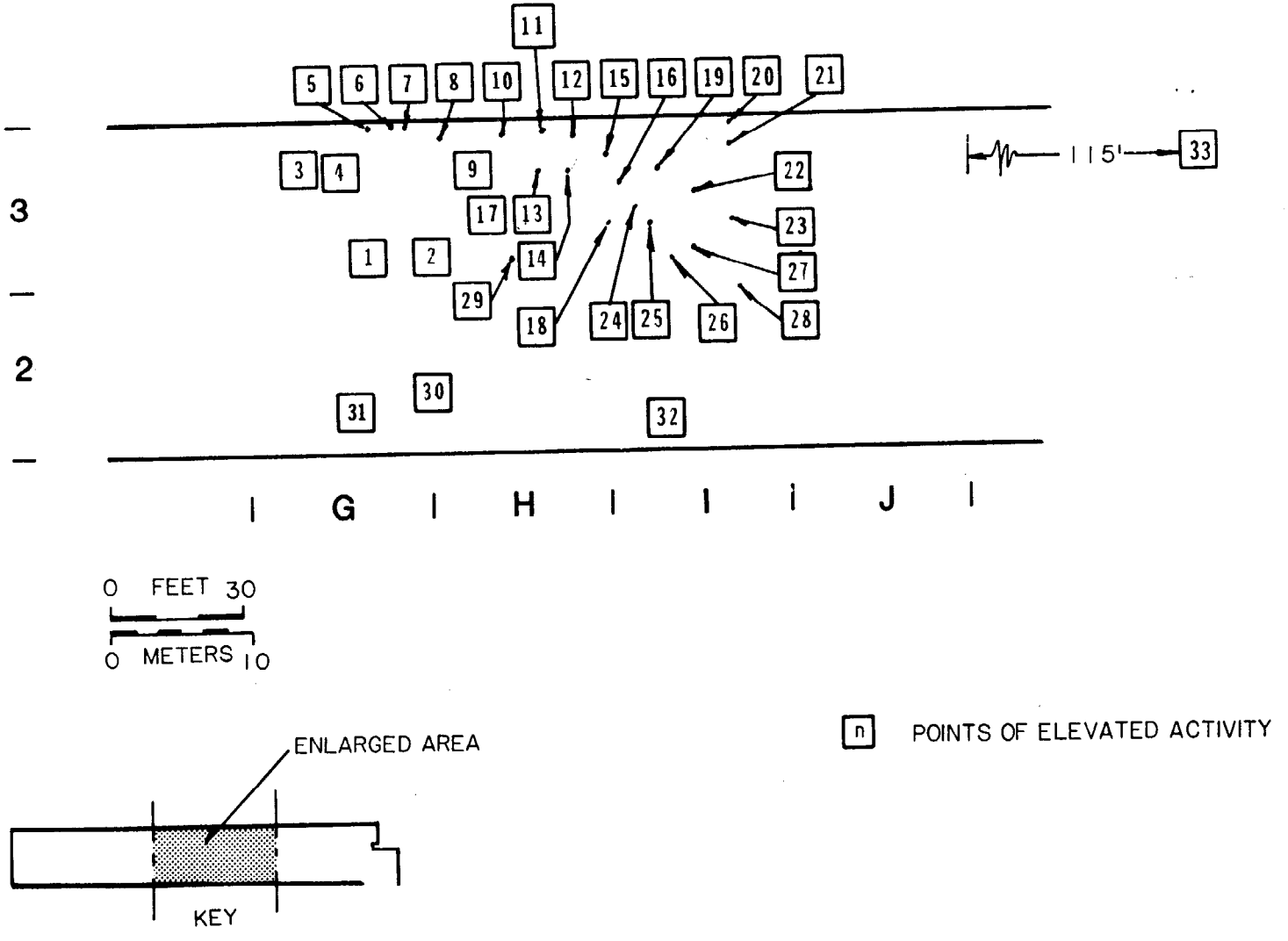

Fig. 3. Site 34 - Survey Plan.
ANL-HP Dwg. No. 82-4

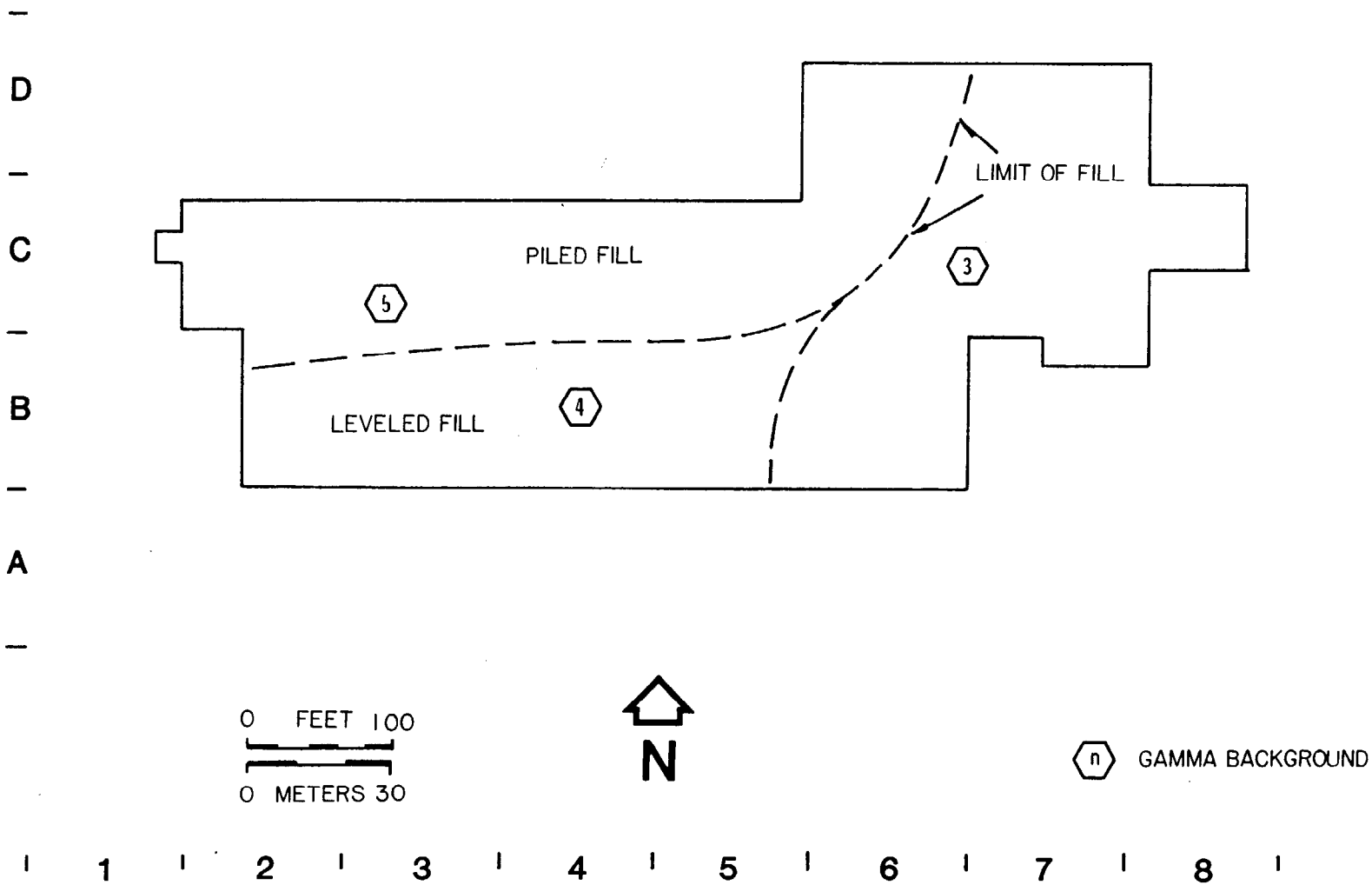

Fig. 4. Site 41 - Survey Plan.
ANL-HP Dwg. No. 83-67

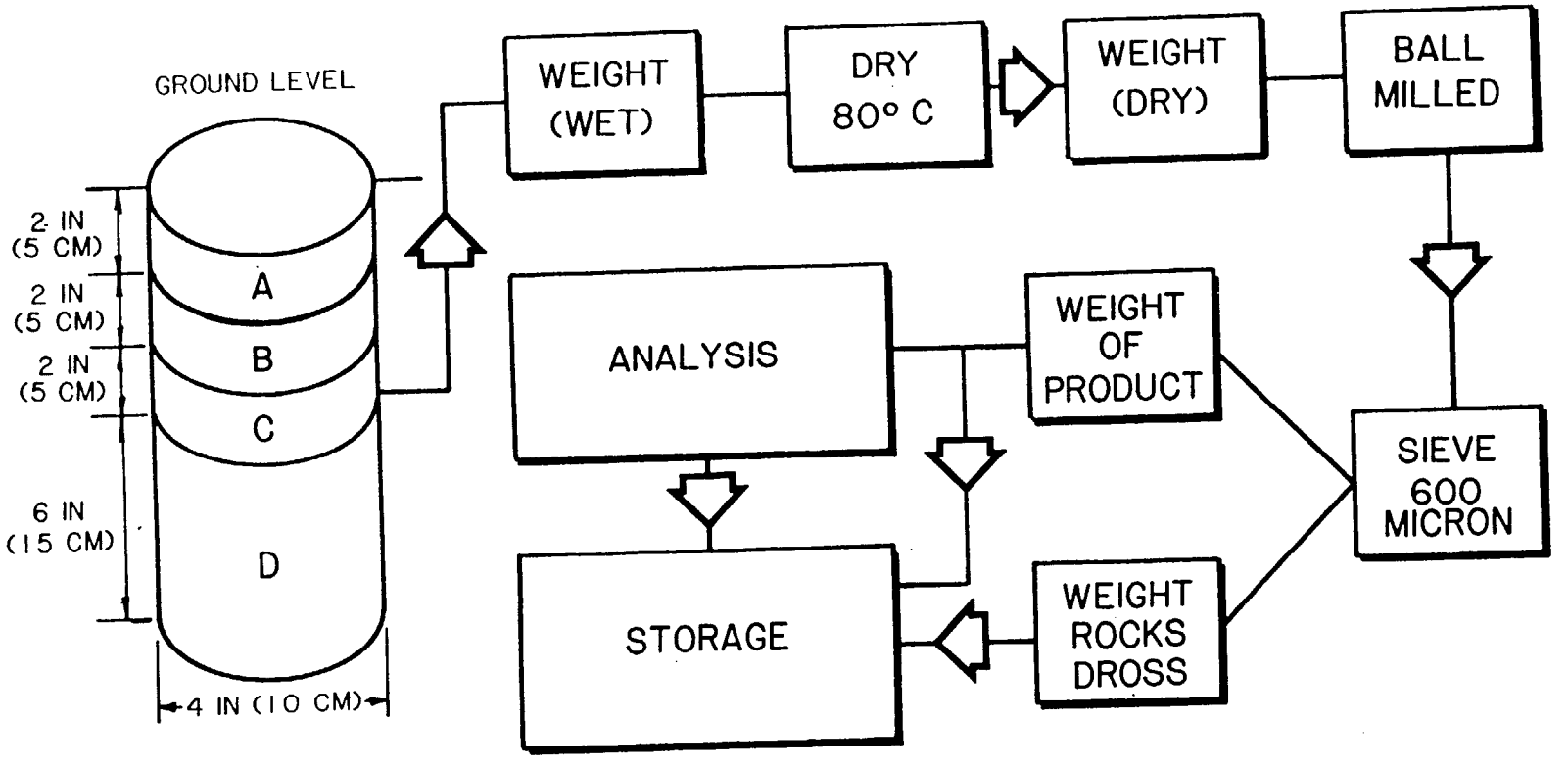

Fig. 5. Environmental Soil Sample Processing.
ANL-HP Dwg. No. 78-2a

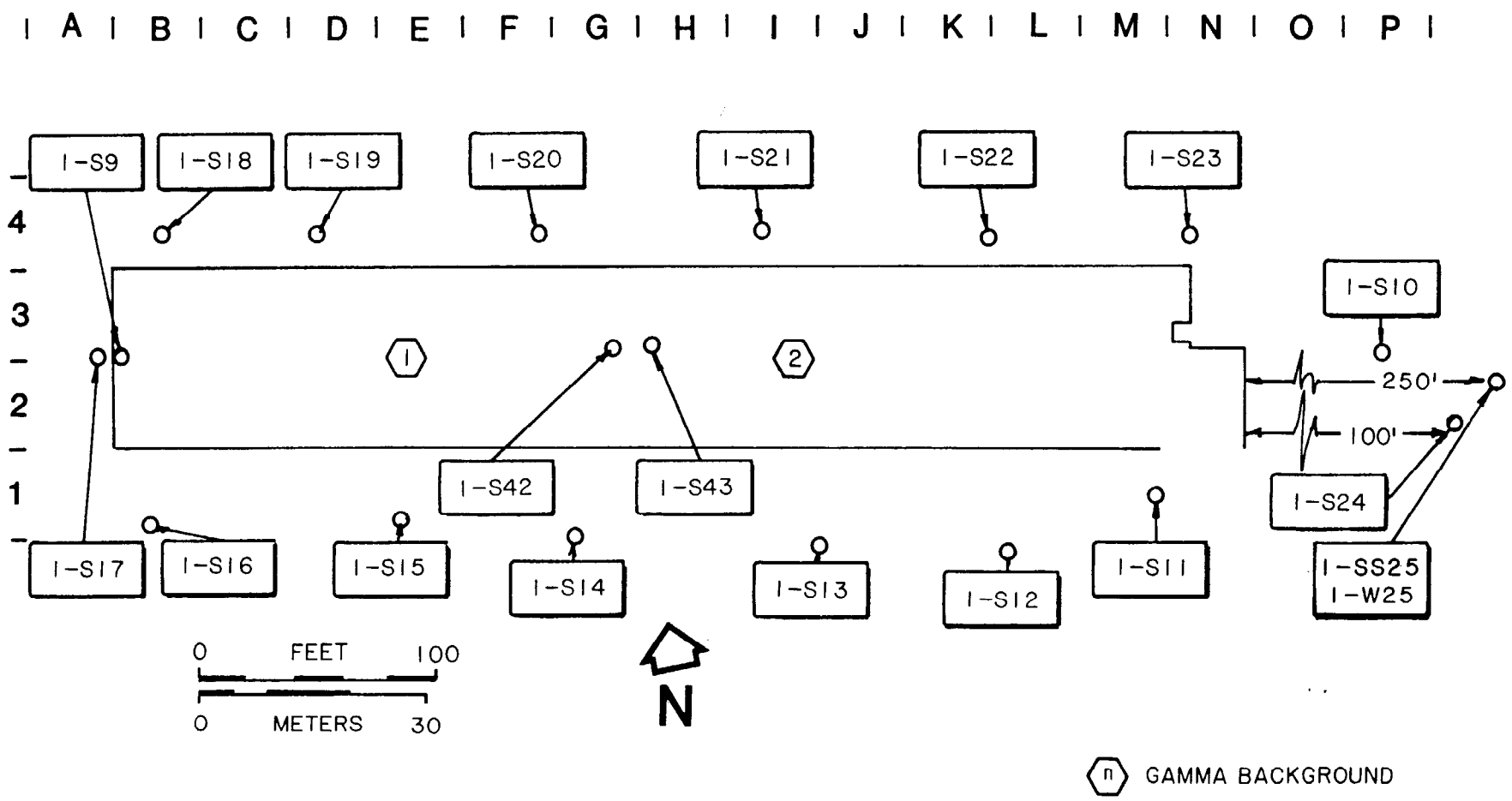

Fig. 6. Site 34 - Soil Coring Locations.
ANL-HP Dwg. No. 82-3

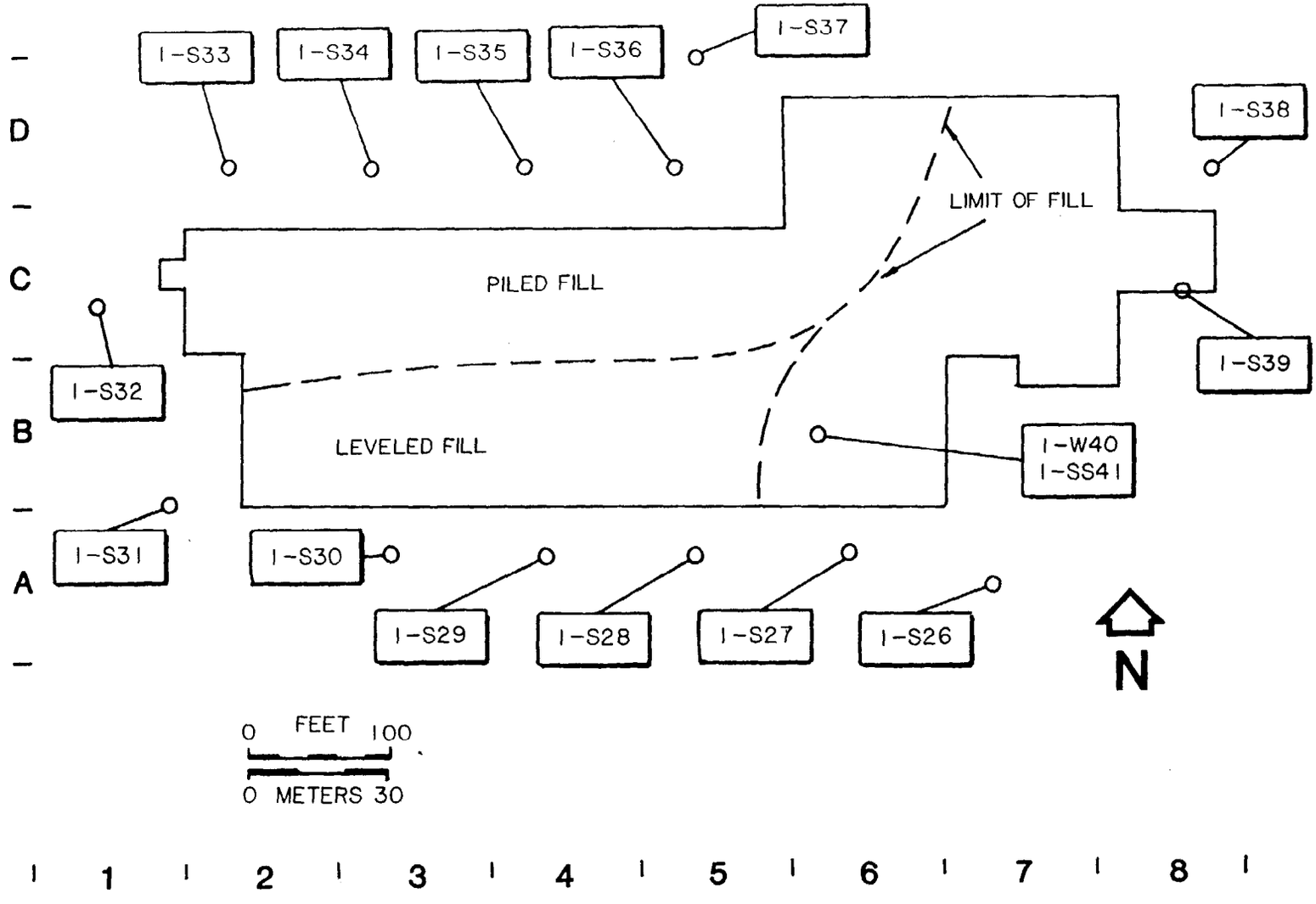

Fig. 7. Site 41 - Soil Coring Locations.
ANL-HP Dwg. No. 82-5.

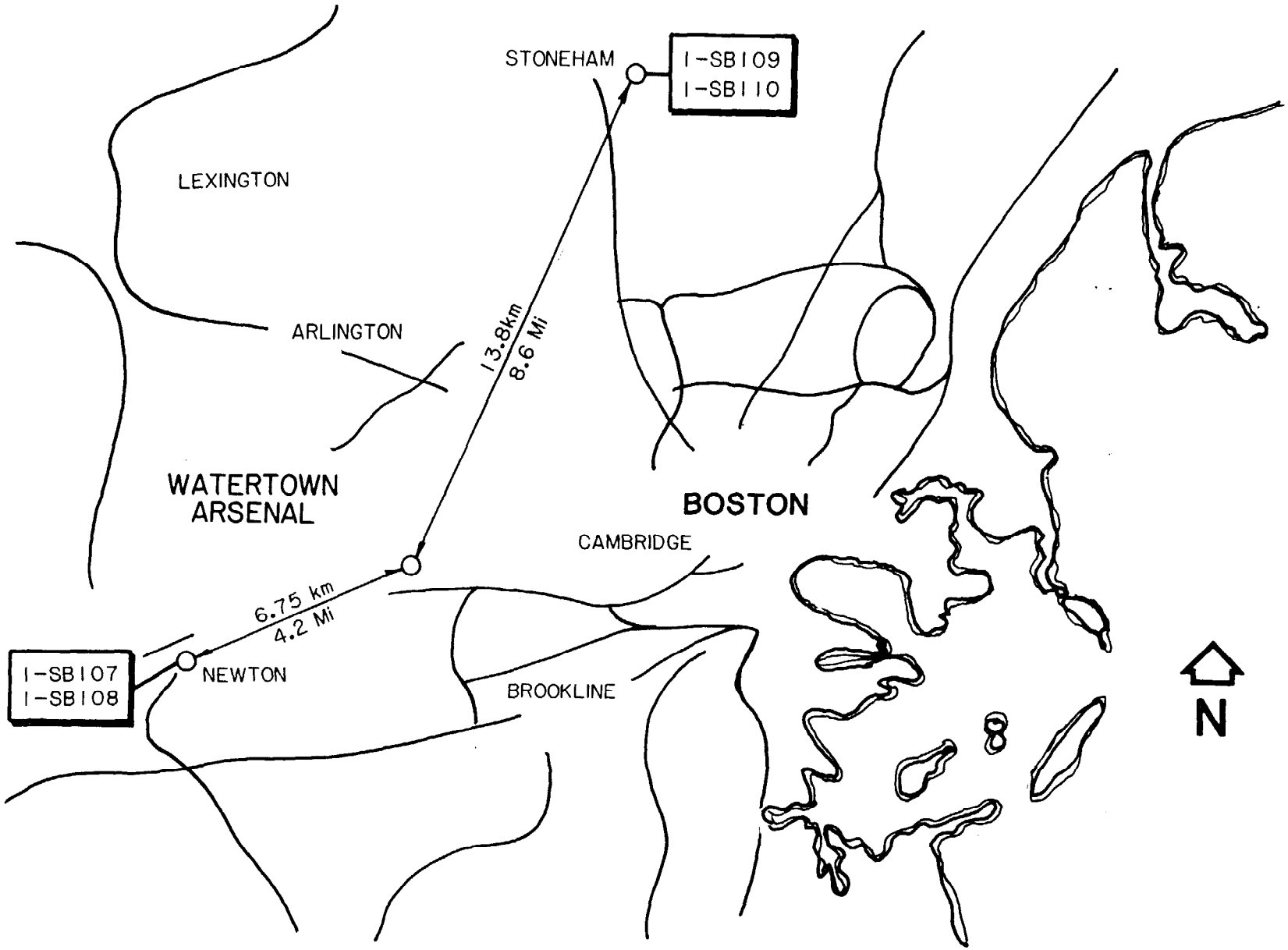

Fig. 8. Background Soil Coring Locations.
ANL-HP Dwg. No. 78-7

TABLE 1
CONTAMINATED AREAS - EXTERIOR SITE 34

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
1, G3				2000	BKGD[c]	0.03	10	1000		< 100 cm²
2, H3				12000	BKGD	0.7	10	10000		< 100 cm²
3, G3				1000	BKGD	0.1	10	3000		< 100 cm²
4, G3				1000	BKGD	0.1	10	2000		10000 cm²
5, G3				1000	BKGD	0.4	10	10000		< 100 cm²
6, G3				3000	BKGD	1.5	10	30000		< 100 cm²
7, G3				5000	BKGD	1.3	10	40000		< 100 cm²
8, H3				5000	BKGD	0.2	10	20000		1400 cm²
9, H3				3000	BKGD	0.03	10	2000		< 100 cm²
10, H3				3000	BKGD	0.03	10	2000		< 100 cm²
11, H3				3000	BKGD	0.03	10	12000		< 100 cm²
12, H3				3000	BKGD	0.03	10	25000		< 100 cm²
13, H3				3000	BKGD	0.07	10	5000		900 cm²
14, H3				3000	BKGD	0.05	10	4000		900 cm²
15, I3				3000	BKGD	0.3	10	15000		900 cm²
16, I3				3000	BKGD	0.2	10	20000		< 100 cm²
17, H3				3000	BKGD	1.2	10	15000		< 100 cm²
18, I3				3000	BKGD	0.03	10	1000		4000 cm²
19, I3				3000	BKGD	7	10	100000		< 100 cm²

TABLE 1
(cont'd.)

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
20, I3				6000	BKGD	0.03	10	2000		7000 cm²
21, I3				3000	BKGD	0.5	10	15000		< 100 cm²
22, I3				2000	BKGD	0.4	10	4000		< 100 cm²
23, I3				3000	BKGD	0.05	10	2000		35000 cm²
24, I3				3000	200	0.2	10	5000		< 100 cm²
25, I3				3000	BKGD	0.3	10	5000		< 100 cm²
26, I3				1000	BKGD	0.3	10	5000		8000 cm²
27, I3				1000	BKGD	0.04	10	2000		< 100 cm²
28, I2				1000	BKGD	0.07	10	3000		22000 cm²
29, H3				2000	BKGD	0.2	10	2000		4000 cm²
30, H2				1000	BKGD	0.3	10	2000		< 100 cm²
31, G2				1000	BKGD	0.2	10	2000		< 100 cm²
32, I2[b]				3000	BKGD	0.6	10	5000		< 100 cm²
33, M3				3000	BKGD	0.03	10	1000		28000 cm²
BKGD[c]				100	0	0.03	10	500		

[a]See Figure 3 for locations.
[b]Metal chips found in area.
[c]BKGD = Background

TABLE 2

AMBIENT RADIATION LEVELS

Location[a]	Sample No.	Length of Measurement (hours)	Radiation Level RSS-111 (μR/h)	Corresponding PRM-7 Measurement (μR/h)
Site 34				
E3	1	3.5	9.2	7-10
I3	2	3.0	9.2	7-10
Site 41				
C6	3	2.3	9.3	7-10
B4	4	3.3	10.1	7-10
C3	5	2.4	9.9	7-10

[a]See Figures 4 and 6 for locations.

TABLE 3

GAMMA SPECTRAL AND URANIUM FLUOROMETRIC ANALYSES OF SOIL SAMPLES[a]

Sample No.	Gamma Spectra (pCi/g±σ) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric[b] μg/g ± 10%	pCi/g ± 10%
			SITE 34		
1-S9	0.03±0.02	4.11±0.28	0.47±0.04	0.7±0.1	0.5±0.1
1-S10-A	1.57±0.11	0.83±0.07	0.98±0.07	7.8±0.8	4.0±0.4[c]
1-S10-B	0.70±0.07	0.98±0.07	1.00±0.07	3.9±0.4	2.4±0.2[c]
1-S10-C	0.40±0.05	0.61±0.07	0.89±0.07	2.5±0.3	1.7±0.2
1-S10-D	0.37±0.05	0.56±0.07	0.96±0.07	2.7±0.3	1.9±0.2
1-S11-A	0.12±0.04	0.79±0.07	1.11±0.08	4.1±0.4	2.6±0.3[c]
1-S11-B	0.80±0.07	0.88±0.07	1.07±0.07	4.8±0.5	2.8±0.3[c]
1-S11-C	0.06±0.03	0.96±0.07	1.27±0.09	13.2±1.3	6.4±0.6[c]
1-S11-D	0.05±0.03	1.03±0.07	1.24±0.09	35.8±3.6	15.5±1.6[c]
1-S12-A	1.15±0.08	0.69±0.07	1.34±0.09	3.7±0.4	2.5±0.3
1-S12-B	0.75±0.07	1.10±0.08	1.23±0.09	4.0±0.4	2.7±0.3
1-S12-C	0.57±0.06	1.10±0.08	1.44±0.10	4.0±0.4	2.7±0.3
1-S12-D	0.16±0.04	1.41±0.10	1.55±0.11	2.6±0.3	1.8±0.2
1-S13-A	1.10±0.08	0.77±0.07	0.86±0.07	2.5±0.3	1.7±0.2
1-S13-B	1.07±0.07	0.54±0.06	0.87±0.07	3.1±0.3	2.1±0.2
1-S13-C	1.11±0.08	1.15±0.08	0.98±0.07	2.5±0.3	1.7±0.2
1-S13-D	0.91±0.07	1.23±0.09	1.42±0.10	3.3±0.3	2.3±0.2
1-S14-A	0.44±0.05	0.95±0.07	1.64±0.11	1.9±0.2	1.3±0.1
1-S14-B	0.87±0.07	0.95±0.07	1.26±0.09	1.8±0.2	1.2±0.1
1-S14-C	0.86±0.07	0.88±0.07	1.39±0.10	1.6±0.2	1.1±0.1
1-S14-D	0.13±0.04	0.87±0.07	1.22±0.09	1.8±0.2	1.2±0.1
1-S15-A	1.40±0.10	0.69±0.07	1.21±0.09	1.9±0.2	1.3±0.1
1-S15-B	1.35±0.09	0.79±0.07	1.34±0.09	1.9±0.2	1.3±0.1
1-S15-C	1.28±0.09	1.17±0.08	1.51±0.11	1.9±0.2	1.3±0.1
1-S15-D	0.31±0.03	1.49±0.11	1.61±0.11	1.7±0.2	1.2±0.1
1-S16-A	0.85±0.08	0.98±0.07	1.17±0.08	2.3±0.2	1.6±0.2
1-S16-B	0.70±0.07	0.90±0.07	1.09±0.07	2.2±0.2	1.5±0.2
1-S16-C	0.59±0.06	0.59±0.06	0.94±0.07	2.2±0.2	1.5±0.2
1-S16-D	0.36±0.04	0.93±0.07	1.11±0.08	2.0±0.2	1.4±0.1
1-S17-A	0.33±0.03	1.19±0.08	1.58±0.11	1.7±0.2	1.2±0.1
1-S17-B	0.20±0.04	0.87±0.07	1.76±0.12	1.7±0.2	1.2±0.1
1-S17-C	0.14±0.04	1.28±0.09	1.86±0.13	1.7±0.2	1.2±0.1
1-S17-D	<0.03	0.95±0.07	1.09±0.08	1.4±0.1	1.0±0.1

TABLE 3
(cont'd.)

Sample No.	Gamma Spectra (pCi/g±σ) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric[b] μg/g ± 10%	pCi/g ± 10%
			Site 34		
1-S18-A	0.59±0.06	0.90±0.07	1.39±0.10	1.9±0.2	1.3±0.1
1-S18-B	0.16±0.04	1.21±0.08	1.41±0.10	2.0±0.2	1.4±0.1
1-S18-C	0.27±0.04	0.98±0.07	1.41±0.10	1.9±0.2	1.3±0.1
1-S18-D	0.07±0.03	1.29±0.09	1.81±0.13	1.8±0.2	1.2±0.1
1-S19-A	1.43±0.10	0.87±0.07	0.95±0.07	2.1±0.2	1.4±0.1
1-S19-B	0.40±0.04	0.34±0.06	0.84±0.07	1.9±0.2	1.3±0.1
1-S19-C	0.27±0.04	0.34±0.06	0.81±0.07	2.1±0.2	1.4±0.1
1-S19-D	0.06±0.03	0.79±0.07	1.08±0.07	1.7±0.2	1.2±0.1
1-S20-A	0.70±0.07	0.94±0.07	1.20±0.09	3.4±0.3	2.3±0.3
1-S20-B	0.52±0.05	1.22±0.09	1.47±0.10	3.3±0.3	2.3±0.2
1-S20-C	0.56±0.05	1.12±0.08	1.87±0.13	2.9±0.3	2.0±0.2
1-S20-D	0.17±0.04	1.55±0.11	1.87±0.13	2.1±0.2	1.4±0.1
1-S21-A	1.47±0.10	0.69±0.07	1.10±0.08	9.5±1.0	4.7±0.5[c]
1-S21-B	1.82±0.13	1.10±0.08	1.47±0.10	12.7±1.3	6.3±0.6[c]
1-S21-C	2.08±0.15	0.80±0.07	1.31±0.09	9.6±1.0	4.9±0.5[c]
1-S21-D	0.43±0.04	1.09±0.08	1.46±0.10	6.1±0.6	3.7±0.4[c]
1-S22-A	0.85±0.07	0.80±0.07	0.77±0.07	2.4±0.2	1.6±0.2
1-S22-B	0.60±0.07	0.54±0.06	0.90±0.07	2.0±0.2	1.4±0.1
1-S22-C	0.34±0.04	0.94±0.07	1.04±0.07	1.8±0.2	1.2±0.1
1-S22-D	0.09±0.04	1.30±0.09	1.02±0.07	2.1±0.2	1.4±0.1
1-S23-A	1.48±0.10	0.80±0.07	1.53±0.11	5.2±0.5	3.4±0.3[c]
1-S23-B	1.29±0.09	1.00±0.07	1.50±0.11	5.4±0.5	3.4±0.3[c]
1-S23-C	1.60±0.11	1.51±0.11	1.37±0.10	5.3±0.5	3.3±0.3[c]
1-S23-D	0.80±0.07	1.01±0.07	1.72±0.12	5.0±0.5	3.4±0.4
1-S24-A	1.77±0.12	0.84±0.07	0.85±0.07	5.7±0.6	3.0±0.3[c]
1-S24-B	1.69±0.12	0.75±0.06	0.81±0.07	8.2±0.8	4.0±0.4[c]
1-S24-C	1.92±0.13	0.50±0.05	1.14±0.08	7.0±0.7	3.8±0.4[c]
1-S24-D	1.06±0.07	0.61±0.05	0.64±0.05	7.1±0.7	3.4±0.4[c]

TABLE 3
(cont'd.)

Sample No.	Gamma Spectra (pCi/g±σ) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric[b] μg/g ± 10%	pCi/g ± 10%
			Site 41		
1-S26-A	0.88±0.07	0.61±0.05	1.41±0.10	1.7±0.2	1.2±0.1
1-S26-B	0.66±0.05	0.81±0.06	1.20±0.08	1.7±0.2	1.2±0.1
1-S26-C	0.33±0.03	0.96±0.07	1.53±0.11	2.4±0.2	1.6±0.2
1-S26-D	0.14±0.04	1.42±0.10	1.53±0.11	2.2±0.2	1.5±0.2
1-S27-A	0.36±0.03	0.30±0.06	1.01±0.07	1.8±0.2	1.2±0.1
1-S27-B	0.54±0.04	0.40±0.04	1.00±0.07	1.6±0.2	1.1±0.1
1-S27-C	0.53±0.04	0.49±0.05	1.15±0.08	1.7±0.2	1.2±0.1
1-S27-D	0.21±0.04	0.90±0.07	0.93±0.07	1.8±0.2	1.2±0.1
1-S28-A	0.32±0.03	0.87±0.07	1.34±0.09	2.7±0.3	1.9±0.2
1-S28-B	0.32±0.03	0.80±0.07	1.36±0.09	3.0±0.3	2.1±0.2
1-S28-C	0.35±0.03	0.88±0.07	1.51±0.11	15.4±1.5	7.5±0.8[c]
1-S28-D	0.18±0.04	1.15±0.08	1.67±0.12	18.1±1.8	8.7±0.9[c]
1-S29-A	0.34±0.03	0.80±0.07	1.51±0.11	2.4±0.2	1.6±0.2
1-S29-B	0.14±0.04	0.92±0.07	1.45±0.10	2.3±0.2	1.6±0.2
1-S29-C	0.12±0.04	1.02±0.07	1.37±0.10	2.5±0.3	1.7±0.2
1-S29-D	<0.03	0.81±0.06	1.04±0.07	1.6±0.2	1.1±0.1
1-S30-A	0.17±0.04	0.34±0.07	0.88±0.07	1.6±0.2	1.1±0.1
1-S30-B	<0.03	0.65±0.05	0.75±0.06	1.9±0.2	1.3±0.1
1-S30-C	0.06±0.03	<0.02	0.87±0.07	2.0±0.2	1.4±0.1
1-S30-D	0.06±0.03	1.05±0.07	1.19±0.08	1.9±0.2	1.3±0.1
1-S31-A	0.24±0.04	0.88±0.07	1.11±0.08	1.9±0.2	1.3±0.1
1-S31-B	0.50±0.04	0.87±0.07	1.23±0.09	3.4±0.3	2.3±0.2
1-S31-C	1.12±0.08	1.06±0.07	1.32±0.09	4.1±0.4	2.8±0.3
1-S31-D	0.19±0.04	1.05±0.07	1.27±0.09	1.9±0.2	1.3±0.1
1-S32-A	2.55±0.18	0.70±0.06	0.99±0.07	3.7±0.4	2.5±0.3
1-S32-B	0.82±0.07	0.91±0.07	1.24±0.09	1.8±0.2	1.2±0.1
1-S32-C	0.71±0.07	1.01±0.07	1.12±0.08	2.1±0.2	1.4±0.1
1-S32-D	0.50±0.04	0.90±0.07	1.28±0.09	1.8±0.2	1.2±0.1
1-S33-A	0.32±0.03	0.84±0.06	1.20±0.08	2.1±0.2	1.4±0.1
1-S33-B	0.36±0.04	0.73±0.05	1.09±0.07	2.5±0.3	1.7±0.2
1-S33-C	0.36±0.04	0.81±0.06	1.12±0.08	2.0±0.2	1.4±0.1
1-S33-D	0.13±0.03	0.54±0.06	0.72±0.05	1.7±0.2	1.2±0.1

TABLE 3
(cont'd.)

Sample No.	Gamma Spectra (pCi/g±σ) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric[b] μg/g ± 10%	pCi/g ± 10%
		SITE 41 (cont'd.)			
1-S34-A	0.14±0.03	1.14±0.07	1.60±0.11	1.6±0.2	1.1±0.1
1-S34-B	<0.03	1.39±0.10	2.08±0.14	2.1±0.2	1.4±0.1
1-S34-C	0.05±0.03	1.20±0.08	1.80±0.13	2.2±0.2	1.5±0.1
1-S34-D	0.11±0.03	1.74±0.12	2.15±0.15	2.8±0.3	1.9±0.2
1-S35-A	0.16±0.04	0.91±0.06	1.42±0.10	1.8±0.2	1.2±0.1
1-S35-B	<0.03	0.99±0.07	1.70±0.12	1.8±0.2	1.2±0.1
1-S35-C	0.11±0.03	0.98±0.07	1.88±0.13	1.9±0.2	1.3±0.1
1-S35-D	0.23±0.03	1.16±0.07	1.98±0.14	1.8±0.2	1.2±0.1
1-S36-A	0.11±0.03	0.70±0.05	1.31±0.09	1.8±0.2	1.2±0.1
1-S36-B	0.09±0.03	1.46±0.10	1.63±0.11	1.8±0.2	1.2±0.1
1-S36-C	0.07±0.03	1.60±0.11	1.85±0.13	1.9±0.2	1.3±0.1
1-S36-D	0.09±0.03	1.54±0.11	1.11±0.08	1.9±0.2	1.3±0.1
1-S37-A	0.13±0.04	0.84±0.06	1.00±0.07	1.7±0.2	1.2±0.1
1-S37-B	0.10±0.03	1.05±0.07	1.25±0.08	1.8±0.2	1.2±0.1
1-S37-C	0.10±0.03	1.04±0.07	1.11±0.08	1.9±0.2	1.3±0.1
1-S37-D	0.09±0.04	1.04±0.07	1.11±0.08	1.8±0.2	1.2±0.1
1-S38-A	0.03±0.02	0.70±0.04	1.04±0.07	0.9±0.1	0.6±0.1
1-S38-B	<0.03	0.70±0.04	0.99±0.07	1.0±0.1	0.7±0.1
1-S38-C	<0.03	0.87±0.06	0.87±0.06	1.2±0.1	0.8±0.1
1-S38-D	0.05±0.03	0.78±0.06	0.87±0.06	1.2±0.1	0.8±0.1
1-S39-A	0.61±0.04	1.21±0.08	1.25±0.08	1.8±0.2	1.2±0.1
1-S39-B	0.73±0.05	1.43±0.10	1.22±0.08	1.7±0.2	1.2±0.1
1-S39-C	0.60±0.04	1.22±0.08	1.54±0.10	1.9±0.2	1.3±0.1
1-S39-D	0.76±0.06	1.78±0.13	1.28±0.09	1.9±0.2	1.3±0.1
		BACKGROUND SOIL ENVIRONMENTAL			
1-SB107-A	1.25±0.06	1.20±0.10	0.96±0.06	3.4±0.5	2.3±0.3
1-SB107-B				4.3±0.6	2.9±0.4
1-SB107-C				4.1±0.6	2.8±0.4
1-SB107-D				3.1±0.4	2.1±0.3
1-SB108-A	0.91±0.05	1.50±0.10	1.00±0.07	2.5±0.5	1.7±0.3
1-SB108-B				6.8±0.6	4.6±0.4
1-SB108-C				3.5±0.5	2.4±0.3
1-SB108-D				3.2±0.4	2.2±0.3

Table 3
(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%
		BACKGROUND SOIL (cont'd.)			
1-SB109-A	2.20±0.10	0.90±0.10	0.68±0.06	2.5±0.4	1.7±0.3
1-SB109-B				2.2±0.4	1.5±0.3
1-SB109-C				1.3±0.4	0.9±0.3
1-SB109-D				2.4±0.4	1.6±0.3
1-SB110-A	<0.05	0.90±0.10	0.72±0.06	4.0±0.4	2.7±0.3
1-SB110-B				17.4±0.8	11.8±0.5
1-SB110-C				5.0±0.4	3.4±0.3
1-SB110-D				3.2±0.4	2.2±0.3

[a]Sample locations are shown in Figures 6, 7, and 8.

[b]Conversion to pCi/g based on Appendix 5.

[c]Conversion factors of 0.405 pCi/μg as calculated for depleted uranium with uranium-236, and 0.683 for natural uranium were used in these cases. The calculation of this conversion factor is presented in Appendix 5. The isotopic composition is taken from the samples presented in ANL-OHS/HP-83-108 from the GSA Site that contain high-levels of depleted uranium with uranium-236 contamination. The data in Table 5 indicates a mixture of natural uranium mixed with the same type of uranium found in the GSA Site. Corrections for the contribution of natural uranium were based on the ^{226}Ra concentration in each sample.

TABLE 4

GAMMA SPECTRAL AND URANIUM FLUOROMETRIC ANALYSES OF WATER AND SLUDGE SAMPLES

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-W25(DS)[a]	< 0.03	0.09±0.05	<0.02	0.00014±10%[a]	0.0001±10%[a]
1-W25(SS)[b]	<0.03	<0.06	<0.02	2.6 ± 0.3	1.8 ± 0.2
1-SS25(DS)	0.21±0.03	<0.06	0.10±0.03	0.001 ±10%[a]	0.0007±10%[a]
1-SS25(SS)	0.85±0.06	0.81±0.06	0.78±0.06	14.9 ± 1.5	10.2 ± 1.0
1-W40(DS)	<0.01	<0.01	0.31±0.03	0.0042 ± 0.0004	
1-W40(SS)	<0.03	<0.06	<0.02	17.4 ±1.7	12.0 ± 1.2
1-SS41(DS)	0.20±0.04	<0.01	<0.01	0.0011 ±0.0001	
1-SS41(SS)	0.81±0.08	0.82±0.08	5.52±0.55	8.5 ±0.8	5.8 ± 0.6

[a]DS corresponds to dissolved solids - (units pCi/mℓ).

[b]SS corresponds to suspended solids - (units pCi/g).

[c]Because of the small sample size, these values were below the detectable limits (BDL).

TABLE 5

URANIUM MASS SPECTROMETRIC ANALYSES
OF SELECTED SOIL SAMPLES

Sample No.	U Isotopes, Atom%				
	233	234	235	236	238
1-S11-D	<0.0003	0.0009	0.2482	0.0043	99.747
1-S28-D	<0.0003	0.0010	0.2557	0.0046	99.739

APPENDIX 1

INSTRUMENTATION

I. PORTABLE RADIATION SURVEY METERS

A. Gas-Flow Proportional Survey Meters

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

Since this instrument has multiple high-voltage settings it can be used to distinguish between alpha and beta-gamma contamination. This instrument was initially used in the beta mode. In the beta mode, the detector responds to alpha and beta paticles and x- and gamma-rays. When areas indicated a higher count rate than the average instrument background, the beta-mode reading was recorded, and the instrument was then switched to the alpha mode to determine any alpha contribution. In the alpha mode, the instrument responds only to particles with high-specific ionization. This instrument is calibrated in the alpha mode with a flat-plate infinitely-thin NBS traceable ^{239}Pu standard, and in the beta mode with a flat-plate infinitely-thin NBS traceable ^{90}Sr-^{90}Y standard. The PAC-4G-3 instruments are calibrated to an apparent 50% detection efficiency. 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/cm^2. 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 ^{226}Ra 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 (2 in) in diameter by 2 mm thick. This instrument is calibrated on three separate discriminators for three energy regions using ^{239}Pu (17 keV), ^{241}Am (59.5 keV) and ^{235}U (185.7 keV) sources. This instrument can be operated in either a differential (to discriminate between different energy regions) or integral mode.

APPENDIX 1
(cont'd.)

D. High Energy Micro "R" Scintillation Survey Meter

An Eberline Micro "R" meter model PRM-7 was used to detect high-energy gamma radiation. This instrument contains an internally mounted NaI(Tℓ) scintillation crystal 2.5 cm (1 in) in diameter by 2.5 cm (1 in) thick and can be used for measuring fields of low-level radiation between 10 μR/h and 5000 μR/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. This instrument has 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. This counter also uses P-10 counting gas. Smears are counted in both the alpha and beta modes. This instrument is 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 sytem and the contamination radionuclides were identified.

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

APPENDIX 2

CONVERSION FACTORS

I. INSTRUMENTATION

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

A. Conversion Factors

	PAC-4G-3		Floor Monitor (FM-4G)	
	Alpha	Beta	Alpha	Beta
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 cm^2
Conversion to 100 cm^2 = 0.31 times Floor Monitor readings

. PAC-4G-3

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

. 2π Internal Gas-Flow Counter, PC counter

Geometry: Solid Steel Spun Top - 0.50

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

APPENDIX 2
(cont'd.)

The PAC-4G-3 and PC counters are calibrated (see Appendix 1) using a flat-plate, infinitely thin ^{226}Ra plus short-lived daughters standard as a source of alpha emissions. The plate was counted in a 2π Internal Gas-Flow Counter (PC counter) with the source leveled to an apparent 2π geometry. The alpha counts per minute (cts/min) reading was found to be 1.9×10^4 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, 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. <u>Conversion Equation (Alpha)</u>

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

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 x bf x sa x 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)

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

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 $\simeq$ 100 minutes)

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

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

Concentration is determined by the equation:

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

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

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

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

t_s = Length of sampling time (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 (^{219}Rn) progeny (AcA, AcB & AcC) decays at the AcB half-life of 36 minutes it cannot be distinguished from the radon (^{222}Rn) progeny using standard air sampling with HV-70 or LB5211 filter media and standard alpha counting techniques. A positive displacement pump is used to collect a sample on millipore (0.5 to 0.8 micron) filter media. The sample rate is approximately 20 liters/minute for a sampling time of at least 90 minutes. The center portion

APPENDIX 3
(cont'd.)

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	$t_s = 40$ min
at	$t = 100$ min	$A = 2000$ dis/min
at	$t = 360$ min	$A = 140$ dis/min
at	$t = 7$ days	$A = 5$ dis/min

For long-lived activity:

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

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

APPENDIX 3
(cont'd.)

For thoron:

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

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

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

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

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

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

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

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

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:

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

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

APPENDIX 3
(cont'd.)

$B_1 = 44$

$B_1/0.84 = 52$ counts

$B_2 = 601$ counts

Actinon = 52/653 = 0.08

Radon = 601/653 = 0.92

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

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

Since we assume that on the average half of the progeny is not adhered to the airborne particulates, the above concentrations are then 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(L) = 0.19 dis/min-m^3

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

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

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

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

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

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

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

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

APPENDIX 4

SAMPLE PREPARATION AND ANALYSIS GENERIC PROTOCOL

I. SOIL-SAMPLE PREPARATION

Soil samples are acquired as previously described. These samples are bagged and identified at the collection site and returned to ANL. If there is an indication of radioactive contamination, the sample is sealed in a Nalgene jar. At ANL, the soil samples are logged into the soil sample book and 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 (approximatley 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 (1¼" x 1¼" Burundum cylinders). The mill jar number is marked on the original container. The jars are sealed and the samples are milled for two hours or until sufficient material is produced to obtain 100 g and 5 g samples for analyses. The samples are milled six at a time. A second set of six jars is prepared while the milling of the first set is proceeding. After each sample is milled, the mill balls are removed with tongs and placed in a tray. A large plastic bag is inverted over the mill jar. Both are inverted and shaken until all the soil is transferred to the bag. If the soil plates the inside of the mill jar, a small paint brush is used to loosen the soil before the jar is inverted. A separate brush is used for each jar to prevent cross-contamination of the soil samples.

After milling, each sample is sieved through a number 30 standard testing sieve (600 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 drawn through the mill jars with a hose connected 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 furance. The first trap is immersed in an ordinary ice bath (0°C); the second trap is immersed in a CO_2-Freon bath (-57°C). The collected tritiated water from both traps is combined, made up to a known volume, and an aliquot taken for liquid scintillation counting of the tritium.

VI. ANALYSIS PROCEDURES

A 100-g fraction from each soil sample is analyzed by high resolution gamma-ray spectroscopy using a germanium crystal detector coupled to a multichannel analyzer. This analysis allows for a quantitative determination of the ^{226}Ra decay chain (via the 609 keV γ-ray of ^{214}Bi) and the ^{232}Th decay chain (via the 908 keV γ-ray of ^{228}Ac as well as any other gamma emitting radionuclide (e.g. ^{137}Cs) present in the soil.

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

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

APPENDIX 4
(cont'd.)

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

A mass spectrometric analysis of the uranium fraction is conducted when it is known or 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 $(\text{wt }\%)_i =$

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

Specific activity for normal uranium:

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

(1.515 dis/min-μg)/(2.22 dis/min-pCi) = 0.683 pCi/μg 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:

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

corresponding to:

(0.9011 dis/min μg)/(2.22 dis/m pCi) = 0.406 pCi/μG U

[a]The 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_a (Note 2) is 2×10^{-13} Ci/m^3 or less or for which the nonoccupational MPC_w (Note 4) is 2×10^{-7} Ci/m^3 or less	^{227}Ac $^{241,242m,243}Am$ $^{249,250,251,252}Cf$ $^{243,244,245,246,247,248}Cm$ $^{125,129}I$ ^{237}Np ^{231}Pa ^{210}Pb $^{238,239,240,242,244}Pu$ $^{226,228}Ra$ $^{228,230}Th$	20	Nondetectable (Note 3)
2	Those nuclides not in Group 1 for which the nonoccupational MPC_a (Note 2) is 1×10^{-12} Ci/m^3 or less for which the nonoccupational MPC_w (Note 4) is 1×10^{-6} Ci/m^3 or less	^{254}Es ^{256}Fm $^{126,131,133}I$ ^{210}Po ^{223}Ra ^{90}Sr ^{232}Th ^{232}U	200	2000 α Nondetectable β,γ (Note 5)
3	Those nuclides not in Group 1 or Group 2		1000	5000

APPENDIX 6
(Cont'd.)

SURFACE CONTAMINATION LIMITS

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

+Disintegrations per minute per square decimeter.

NOTES:

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

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

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

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

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

APPENDIX 6
(Cont'd.)

TABLE 2
ALTERNATE SURFACE CONTAMINATION LIMITS

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

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

APPENDIX 6
(Cont'd.)

ALTERNATE SURFACE CONTAMINATION LIMITS

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

+Disintegrations per minute per square decimeter.

NOTES:

(1) U_{nat} and decay products.

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

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

APPENDIX 6
(Cont'd.)

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

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

(These have been retyped for purposes of this report)

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

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

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

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

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

   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 (nuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above.	5000 dis/min-100 cm^2 $\beta\gamma$	15,000 dis/min-100 cm^2 $\beta\gamma$	1000 dis/min-100 cm^2 $\beta\gamma$

APPENDIX 6
(Cont'd.)

TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LEVELS

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

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

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

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

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

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

APPENDIX 6
(Cont'd.)

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

712.1 Purpose

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

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

712.2 Scope

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

712.3 Definitions

As used in this part:

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

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

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

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

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

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

APPENDIX 6
(Cont'd.)

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

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

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

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

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

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

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

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

712.4 Interpretations

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

712.5 Communications

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

APPENDIX 6
(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.

APPENDIX 6
(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.

APPENDIX 6
(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.

APPENDIX 6
(cont'd.)

IV. 40 CFR Part 192

HEALTH AND ENVIRONMENTAL PROTECTION STANDARDS
FOR
URANIUM MILL TAILINGS

(Excerpts have been retyped for purposes of this report.)

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, escept in the case of a site at or near Slick Rock, Colorado, unless--

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

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

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

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

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

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.

APPENDIX 6
(cont'd.)

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

192.12 Standards

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

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

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

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

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

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

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

APPENDIX 6
(cont'd.)

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

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

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

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

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

APPENDIX 6
(cont'd.)

VI. EXCERPTS FROM DOE 5480.1, Chapter XI

"Requirements for Radiation Protection"

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

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

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

CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

			Table I Controlled Area		Table II Uncontrolled Area	
Element (atomic number)	Isotope	soluble (S) insoluble (I)	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:

Volume (Average) = 34,800 ft^2 (area) x 3.6 ft (avg. depth) = 125,000 ft^3
= 3,550 m^3

Volume (Maximum) = 34,800 ft^2 (area) x 9 ft (max. depth) = 314,000 ft^3
= 8,880 m^3

Mass (Average) = 3,550 m^3 x 1,500 kg/m^3 = 5.33×10^6 kg
Mass (Maximum) = 8,880 m^3 x 1,500 kg/m^3 = 1.33×10^7 kg

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

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

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

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

APPENDIX 8

EVALUATION OF RADIATION EXPOSURES

INTRODUCTION

A. Types of Radiation

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

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

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

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

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

B. Sources of Radiation

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

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

APPENDIX 8
(cont'd.)

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.

APPENDIX 8
(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.

APPENDIX 8
(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

APPENDIX 8
(cont'd.)

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

EVALUATION OF RADIATION DOSE AND POTENTIAL HAZARD

Radiation, regardless of its sources, is considered to be a hazard because of its potential for producing adverse effects on human life. Very large amounts of radiation received over a brief period, i.e., hundreds of rem delivered within a few hours, can produce severe injury or death within days or weeks. Distributed over longer intervals, however, these same doses would not cause early illness or fatality. At 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 lead to estimates of the risk of radiation-induced cancer; estimated risks for the lower doses have been derived by linear extrapolation from the higher doses. All radiation exposures then, no matter how small, are assumed to be capable of increasing an individual's risk of contracting cancer.

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

Estimates of health effects from radiation doses are usually based on risk factors as provided in 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

APPENDIX 8
(cont'd.)

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

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

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

APPENDIX 8
(cont'd.)

DOSE DETERMINATION CALCULATION

External Exposure

External penetrating radiation dose rates are measured on contact with an end-window beta-gamma Geiger-Mueller (GM) detector (7 mg/cm^2 window), and at 1 meter with a NaI crystal detector (1 in diameter by 1 in thick) manufactured by Eberline Corporation (PRM-7 μR 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 m^3/working day and a soil density of 1.5 g/cm^3, 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 ^{238}U, ^{235}U, ^{234}U and ^{232}Th (and all their significant daughters), as determined in Reference 2, are given in Table 1. Normal uranium is assumed to be composed of 2.26% ^{235}U, and 97.74% ^{234}U and ^{238}U (in equilibrium) by activity.

An example calculation based on the above scenarios assumes a soil contamination level of 5 pCi/g of radium-226. This gives the following dose levels (50-year dose commitment):

APPENDIX 8
(cont'd.)

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

Bone: 6.5×10^{-2} mrem/pCi x 5 pCi/g x 100 g = 32.5 mrem
Total Body: 5.5×10^{-3} mrem/pCi x 5 pCi/g x 100 g = 2.75 mrem

Inhalation (rototilling and breathing ensuing aerosol):

Lung: 1.1×10^{-1} mrem/pCi x 2.285 pCi/m^3 x 9.6 m^3 = 2.4 mrem
Bone: 9.2×10^{-2} mrem/pCi x 2.285 pCi/m^3 x 9.6 m^3 = 2.0 mrem
Total Body: 9.5×10^{-3} mrem/pCi x 2.285 pCi/m^3 x 9.6 m^3 = 0.2 mrem

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}

[a]Data taken from Reference 2.

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

APPENDIX 8
(cont'd.)

REFERENCES

1. U.S. Department of Energy 1981. "Requirements for Radiation Protection" DOE 5480.1, Chapter XI.

2. D. E. Dunning, Jr., et al. 1981. "Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities, Volume III." ORNL/NUREG/TM-190/V3. Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission.

Distribution for DOE/EV-0005/37 (ANL-OHS/HP-83-106)

Internal:

E. S. Beckjord
H. Drucker
J. D. DePue
J. G. Ello
K. F. Flynn
C. A. Hunckler
A. L. Justus
J. H. Kittel
N. D. Kretz
D. P. O'Neil
M. J. Robinet
C. M. Sholeen
W. H. Smith
J. P. Unik
V. R. Veluri
R. A. Wynveen
OHS/HP Publications File (10)
ANL Patent Dept.
ANL Contract File
ANL Libraries (4)
TIS Files (6)

External:

DOE-TIC, for distribution per UC-70A (123)
H. J. Rauch, Manager, Chicago Operations Office, DOE (10)
J. E. Baublitz, Director, Remedial Action Projects, DOE (5)
D. E. Patterson, Director, Office of Operational Safety, DOE (5)
C. E. Miller, Jr., Director, Surplus Facilities Management Program, DOE-RL (5)
T. E. Dabrowski, UNC Nuclear Industries, Richland (5)