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DOE/EV-0005/35

ANL-OHS/HP-83-103

FORMERLY UTILIZED MED/AEC SITES REMEDIAL ACTION PROGRAM

RADIOLOGICAL SURVEY OF CHEMICALS GROUP, OLIN CORPORATION (Formerly Blockson Chemical Company) JOLIET, ILLINOIS

March 27-November 28, 1978

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

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

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

NTIS price codes
Printed copy: A06
Microfiche copy: A01

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

DOE/EV-0005/35
ANL-OHS/HP-83-103

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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RADIOLOGICAL SURVEY OF CHEMICALS GROUP, OLIN CORPORATION (formerly Blockson Chemical Company) JOLIET, ILLINOIS

March 27—November 28, 1978

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May 1983

Work Performed under

Budget Activity ERDA RK-01-05-02-3
and ANL 73706

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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 used by the Manhattan Engineer District (MED) and the AEC for work involving the handling of radioactive materials.* Since the early 1940's, the control of over 100 sites that were no longer required for nuclear programs has been returned to private industry or to 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 Chemicals Group, Olin Corporation (formerly Blockson Chemical Company) in Joliet, Illinois, was one such site.

Under contract to the AEC, the Blockson Chemical Company developed and operated a uranium-recovery facility to extract uranium from wet-process phosphoric acid. The phosphoric acid was an intermediary in Blockson Chemical's commercial manufacture of fertilizers and phosphate chemicals from phosphate rock. The phosphate rock was of Florida origin and typically contained about 42 pCi ^{226}Ra/g (Ref. 1) and 86 pCi U/g (Appendix 5). The AEC work was conducted within a 30.5-m by 53.3-m (100 ft x 175-ft)** brick structure (Building 55) presently used for the chemical processing of phosphoric acid.

To determine if any radioactive contamination remains as a result of the AEC activities, a comprehensive radiological assessment of Building 55 and its

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*The various types and sources of radiation mentioned in this report are discussed in more detail in Appendix 8.

**When metric units are followed (in parentheses) by English units, the measurements were originally made in English units and then converted into metric. In cases where only metric units are given, the values were either originally given in metric, or resulted from calculations involving numbers previously converted from English into metric.

grounds was conducted on an intermittent basis from March 27 to November 28, 1978. The assessment included instrument and smear surveys and analysis of samples of air, soil, and other materials. Direct instrument surveys and smear surveys indicated that some contamination was present throughout the building, mainly on the concrete floors, the overhead beams, and on the mixing vats, Kelly filtration devices, and processing tanks. Contamination was found at 47 spots or localized areas and 11 larger, general areas throughout the building and on the roof. Through gamma-spectral analysis, the contaminants were identified as uranium and radium. The contamination may be due to past AEC activities or to the present use of the building for the chemical processing of phosphate products from ground phosphate rock from Florida.

The beta-gamma readings obtained with a gas-flow proportional survey meter at the contaminated areas ranged from 3.4×10^2 to 1.4×10^6 dis/min-100 cm^2. The alpha readings at these locations ranged from background to 5.8×10^3 dis/min-100 cm^2. Thirty-two of the areas gave Geiger-Mueller (GM) End Window exposure readings at contact ranging from just distinguishable from background to 7 mR/h. Seven GM End Window readings taken at 1 m (3 ft) were distinguishable from background, the highest being 0.2 mR/h.

Radioactivity was detected on 15 smears, indicating the presence of loose contamination. Beta-gamma levels on the smears ranged from background to 850 dis/min-100 cm^2, and alpha levels were from background to 640 dis/min-100 cm^2.

The radiation readings obtained in the contaminated areas were compared with standards and guidelines in the American National Standard Institute Draft N13.12, "Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use," and in the Nuclear Regulatory Commission's "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material."

Normal uranium was identified as the principal contaminant within the building; on the roof, normal uranium and the radium decay-chain contamination were identified. (The roof was the only area where radium was identified as being among the contaminants.) Therefore, the Draft ANSI Standard and NRC

limits for uranium were used as the basis of comparison for readings within the building and the limits for radium were used as the basis of comparison for the readings taken on the roof. Thirty-three spots or localized areas and three general areas within Building 55 exceeded the acceptable surface contamination limits for uranium, and the general roof area exceeded the acceptable limits for radium-226. Additionally, two spots or localized areas on the roof substantially exceeded those limits. Eleven areas of contamination within the building also exceeded NRC Guidelines for maximum radiation levels resulting from surface contamination with beta-gamma emitters.

The radon decay-product concentrations measured in air samples collected at selected locations in the building, including the areas where contamination was found, ranged from 0.0014 to 0.0061 Working Levels (WL), including background. Under the Surgeon General's Guidelines, no need for remedial action is indicated when indoor concentrations of radon daughters are less than 0.01 WL above background. The radon-daughter concentrations measured in Building 55 were within normally expected background concentrations. No long-lived radionuclides were detected in any air sample.

Ten soil samples were collected about the grounds of Building 55 to test for the presence of any radionuclides that could have been spilled or released outside the building. Analyses of the soil samples showed uranium concentrations ranging from 0±0.2 to 87.7± 4.8 pCi/g. All four segments of each of two soil samples collected just south and east of the building contained elevated levels of uranium (16±1 to 87.7± 4.8 pCi/g). In comparison, background samples taken from the Chicago area indicated concentrations of natural uranium ranging from 0.6±0.1 to 2.2±0.3 pCi/g (see Table 6). There now are no standards specifying a limit for uranium in soil, but the measured uranium concentrations in the top segment of each of the two samples with elevated levels were in excess of the proposed interim soil limit of 40 pCi/g for decommissioning and decontamination projects (Ref. 2 and Appendix 6). The two samples also contained elevated levels of radium (7.96±0.40 and 18.1±0.91 pCi/g), in excess of the 5 pCi/g limit as specified in the Environmental Protection Agency's 40 CFR 192, Subpart B ("Standards for Cleanup of Open Lands and Buildings Contaminated with Residual Radioactive Materials from Inactive Uranium Processing Sites"). These elevated levels were judged by the survey team to be a result of the spilling or dumping of some residue from the chemical process.

To evaluate the radiation exposure potential, potential doses resulting from radiation exposure were calculated for pathways that could result in the maximum dose from external penetrating radiation and the maximum doses from inhalation/ingestion of the residual radioactive material detected in the survey. The calculations were based on reasonably conservative assumptions and, therefore, most probably overestimate the potential dose. The maximum potential external penetrating radiation dose was calculated to be 340 mrem per year. This dose is equivalent to an increase of about 410% over the 82-mrem natural background whole-body dose and is about 68% of the 500-mrem standard for an individual in an uncontrolled area. The internal radiation 50-year dose commitments from potential inhalation/ ingestion of contamination were calculated to be 8300 mrem to the lung, 120 mrem to the bone, 26 mrem to the kidney, and 250 mrem whole body. For the lung, bone, and kidney, these doses represent additions of 4600%, 70%, and 32% to the 179-mrem, 171-mrem, and 82-mrem annual natural background lung, bone, and kidney doses, respectively, and are 55%, 8% and 1.7% of the 1500-mrem annual standard for an individual in an uncontrolled area. For the whole body, the calculated dose represents an increase of 300% to the 82 mrem annual natural background whole-body dose and is 50% of the 500-mrem annual standard for an individual in an uncontrolled area. Very few individuals are expected to acquire such doses or dose commitments annually.

In order to reduce the potential for radiation exposure, remedial measures such as stablization of the contamination in place would be applicable as a short-term measure. To reduce the risk in the event that building modifications take place in the future, health physics procedures and coverage are recommended. The long-term solution would involve decontamination by removal of the radioactive residues from the areas in the facility where contamination was detected.

This radiological assessment was performed by the following Health Physics personnel of the Occupational Health and Safety Division, Argonne National Laboratory, Argonne, Illinois: R. A. Wynveen, W. H. Smith, C. B. Mayes, P. C. Gray, and D. W. Reilly.

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RADIOLOGICAL SURVEY OF CHEMICALS GROUP, OLIN CORPORATION (formerly Blockson Chemical Company) JOLIET, ILLINOIS

ABSTRACT

A comprehensive radiological survey was conducted at Building 55 of the Chemicals Group, Olin Corporation, Joliet, Illinois. Work performed here for the AEC included the operation of a uranium-recovery facility for the extraction of uranium from wet-process phosphoric acid. At the time of this survey, the building was still used for the chemical processing of phosphate products from ground phosphate rock of Florida origin.

The survey included measurements of alpha and beta-gamma contamination, both fixed and removable; beta-gamma exposure readings at contact and at 1 m (3 ft) above the floor or ground level; estimates of radon-daughter concentrations in the air as airborne contaminants; and determinations of concentrations of ^{137}Cs, the ^{232}Th decay chain, the ^{226}Ra decay chain, and uranium in the soil on the site.

Thirty-three spots or localized areas and three larger general areas within Building 55 exceeded the allowable limits as given in the Draft ANSI Standard N13.12 for uranium, and the general roof area exceeded the acceptable limits for radium-226. Additionally, two spots or localized areas on the roof substantially exceeded those limits. In 15 instances, the contamination was found to be removable from surfaces and readily available for transfer to other locations.

Concentrations of radon daughters in the air of the building, as measured by grab-sampling techniques, were below the limit of 0.01 WL above background as given in the Surgeon General's Guidelines. Calculated radon concentrations based on the radon-daughter determinations were below the Concentration Guide for ^{222}Rn in uncontrolled areas (3 pCi/ℓ) as stated in the DOE document "Requirements for Radiation Protection" (DOE 5480.1, Chg. 6). No long-lived radionuclides were detected in any air sample.

Analyses of ten soil samples from the site indicated significantly elevated concentrations of uranium and radium at two sampling locations near Building 55. The levels of uranium were in excess of the 40 pCi/g interim limit proposed in the report "Interim Soil Limits for D&D Projects"; the levels of radium were in

excess of the 5 pCi/g limit as specified in the EPA's Cleanup Standards for Inactive Uranium Processing Sites.

Based on a hypothetical exposure scenario, the maximum potential dose equivalent to an individual for external penetrating radiation resulting from exposure to radioactivity at this site was calculated to be 340 mrem per year. This represents an increase of about 410% above the 82-mrem annual natural background whole-body dose and is 68% of the 500-mrem standard for an individual in an uncontrolled area. The internal radiation 50-year dose commitments from potential inhalation/ingestion of radioactive material were calculated to be 8300 mrem to the lung, 120 mrem to the bone, 26 mrem to the kidney, and 250 mrem whole body. For the lung, bone, and kidney, these doses represent additions of 4600%, 70%, and 32% to the 179-mrem, 171-mrem, and 82-mrem annual natural background lung, bone, and kidney doses, respectively, and 550%, 8%, and 1.7% of the 1500-mrem annual standard for an individual in an uncontrolled area. Few individuals are expected to acquire such doses or dose commitments annually.

In order to reduce the potential for radiation exposure, remedial measures such as stabilization of the contamination in place would be applicable as a short-term measure. To reduce the risk in the event that building modifications take place in the future, health physics procedures and coverage are recommended. The long-term solution would involve decontamination by removal of the radioactive residues from the areas in the facility where contamination was detected.

INTRODUCTION

A radiological survey was conducted of Building 55 at the Chemicals Group, Olin Corporation Joliet Plant (formerly Blockson Chemical Company). The plant is located south of Joliet, Illinois, between Patterson Road on the east and the Des Plaines River on the west (see Fig. 11).

In 1951, the Blockson Chemical Company entered into a contract with the Atomic Energy Commission to conduct a development program for the extraction of uranium from wet-process phosphoric acid. The phosphoric acid was produced by Blockson Chemical as an intermediate product in its normal commercial operations involving the processing of phosphate rock in the manufacture of fertilizers and phosphate chemicals. Recovery of uranium by this process had the advantage that mining and leaching costs were borne by the phosphate products. Phosphate rock

throughout the world contains uranium in concentrations ranging from a few picocuries to a few hundred picocuries per gram. The uranium decay products in the ores, at least through ^{226}Ra, have generally been found to be in secular equilibrium. The major occurrences of uranium in phosphate minerals in the United States are the phosphate rock deposits of marine origin of Florida and Idaho. The content of these deposits ranges from 35 to 200 pCi U/g (Ref. 3) and 17 to 100 pCi $^{226}Ra/g$ (Appendix 5). The phosphate rock used by Blockson Chemical was from Florida and typically contained about 42 pCi $^{226}Ra/g$ (Ref. 1) and 86 pCi U/g (Appendix 5). In the Blockson process for the recovery of uranium from phosphoric acid, the phosphate rock was calcined prior to digestion with sulfuric acid and subsequent conversion to monosodium phosphate. Details of the digestion and subsequent conversion process are both proprietary. An aqueous solution of monosodium phosphate was then chemically reduced to precipitate a uranous-phosphate. The precipitate of uranous phosphate, containing 40% to 60% U_3O_8, was then dried for shipping (Ref. 4).

The AEC work was conducted in Building 55, a 30.5-m by 53.3-m (100 x 175-ft) brick structure. The contract included the operation of a uranium-recovery facility, and production was ultimately limited to not more than 50,000 pounds (25 tons) of U_3O_8 in uranium concentrate per year (Ref. 5). In 1955, Blockson Chemical Company was sold to Olin Mathieson Chemical Corporation, now known as Chemicals Group, Olin Corporation. The uranium-recovery work was completed in 1962 when the AEC contract expired. At the time of this survey, Building 55 was used for the chemical processing of phosphate products from Florida phosphate rock.

No reports could be found of radiation* surveys or decontamination efforts being conducted at Building 55 after the termination of AEC activities. It was, therefore, specified that a radiation survey should be undertaken to determine if any detectable radioactive contamination remains as a result of the AEC operations. The radiological survey was performed on an intermittent basis from March 27 to November 28, 1978.

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*See Appendix 8 for a detailed discussion and definitions of the various terms and concepts mentioned in this report relative to types of radiation, exposure, doses, and similar topics.

SURVEY AND ANALYTICAL TECHNIQUES

General

Portable radiation survey instruments were used to perform a radiological survey of all accessible floors and original interior wall surfaces to a height of 2 m (7 ft). A representative selection of accessible overhead structures such as beams, pipes, vents, and light fixtures was also surveyed. Surveys were performed in certain of the tanks, mixing vats, and Kelly filtration devices in the building. Some original surfaces might have been painted or otherwise covered since the AEC era. Even though not the original surfaces, such areas were surveyed with instruments that have some capability to detect beta-gamma activity on the original, underlying surfaces. The locations of accessible areas surveyed are listed in Table 1 and shown in Figures 1 through 9.

Instrumentation Used for Direct Survey

Three types of portable survey instruments were used in the direct surveys. An Eberline gas-flow proportional probe (FM-4G) with a detection area of 325 cm^2 and using the Eberline PAC-4G-3 electronics was used to survey the floors. A PAC-4G-3 with a hand-held gas-flow proportional probe, and with a detection area of 51 cm^2, was used to survey the walls and other areas inaccessible with the FM-4G. An Eberline Model 530 Geiger-Mueller (GM) detector with an Eberline HP-190 end-window probe was used to measure the contact exposure rate (mR/h) of all contaminated areas. This instrument was held 1 m (3 ft) above the floor to determine general ambient background radiation levels throughout the surveyed area. The instruments are described in more detail in Appendix 1.

Although ^{239}Pu and ^{90}Sr-^{90}Y standards were used to calibrate the gas-flow instruments, it should be noted that the numerous isotopes that could be encountered exhibit emission energies differing from those of the standards used in the calibration. When detecting known isotopes that emit alpha and beta energies differing from those of the standards, such as normal uranium, a conversion factor for those particular radionuclides was developed to determine the appropriate yield. (The methods used to determine the conversion factors are

described in Appendix 2.) All readings of disintegrations per minute (dis/min) per 100 cm^2, as reported in Table 1, are equated to normal uranium, unless otherwise stated. It should also be noted that since calibrations are to infinitely-thin flat-plate standards, all readings as reported should be regarded as minimal values. No corrections were made for absorption by surface media.

When possible, the contaminant radionuclides were identified by performing a gamma-spectral analysis on a contaminated item or on a sample of material taken from a contaminated area.* A multichannel analyzer with a 7.6-cm by 7.6-cm NaI(Tℓ) crystal (described in Appendix 1) was used for this purpose. This instrument, along with all other survey and sampling devices, was housed in a mobile laboratory, a specially designed, converted motor home.

Smear Surveys

Dry smears were taken at representative locations throughout the entire building. Smears were taken on original structures and components such as walls, floors, pipes, and vents. All smears were taken with Whatman No. 1 filter paper, 4.25 cm in diamter. A standard smear is obtained by applying moderate pressure by the tips of the first two fingers to the back of the filter paper while rubbing the paper over the surface. Smears of about 900 cm^2 (1 ft^2) were normally taken. A smear of 100 cm^2 was taken if an area or object had an instrument reading higher than the expected normal background, or if there was excessive dirt or dust in an area.

Two different instruments were used to measure (count) the contamination on the smears. They were first counted in groups of ten using a 10-wire flat-plate gas-flow proportional detector developed by ANL. The instrument detects alpha and beta particles and x- and gamma-rays. Additionally, at least one smear of each group was removed and counted in the more sensitive Nuclear Measurements Corporation 2π Internal Gas-Flow Proportional Counter (PC counter) using an aluminized Mylar cover (Mylar spun top) over the smear. All smears from areas or objects with elevated direct readings and smears in groups indicating readings above the instrument background levels in the 10-wire assembly were indi-

*Such analysis was performed on three samples collected during this survey.

vidually counted in the PC counter. Smears were counted in each detector for both alpha and beta-gamma activity. These instruments are described in Appendix 1.

The factors used to convert instrument counts to disintegrations of a particular isotope for all the smears are given in Appendix 2. Unless otherwise indicated, all contamination on the smears reported in Table 1 is equated to normal uranium, as described in Appendix 2.

The results of the instrument and smear surveys are given in Table 1, and the locations of elevated instrument readings and smear locations are shown in Figures 1 through 9.

Air Samples

Air samples were collected with a commercial vacuum cleaner modified at ANL for use as a particulate air-sampling device. A flow rate of 40 cubic meters per hour (m^3/h) was used. A 10% portion (5 cm in diameter) was removed from the filter media after collection and counted for both alpha and beta-gamma activity in the PC counter, using a Mylar spun top. The counting results were used to determine radon and radon-daughter concentrations and the presence of any long-lived radionuclides. Information and assumptions used to determine the radon daughter concentrations are presented in Appendix 3; the results are given in Table 2; and the locations where air samples were collected are shown in Figures 1 through 5.

Soil and Material Samples

Environmental soil corings were collected at 10 selected locations outside the building to detect any radioactive material that might have been spilled or released. The sampling locations are shown in Figure 10. Uranium-fluorometric and gamma-spectral analyses were conducted on these soil samples. The corings were taken with a 10 cm (4 in) diameter, 15 cm (6 in) long, right-circular cylinder cutting tool commonly used to cut golf-green holes. Each core was 30 cm long, and each was divided into four segments. Starting from the surface, three separate 5-cm segments were cut, bagged, and marked A, B, and C, respectively; the final segment of 15 cm was marked D (see Fig. 12).

The segmented coring technique was used to determine if any contaminant migration had occurred, to reduce the dilution of upper-level soil with the lower-level segments with respect to the surface deposition of the contaminants (or vice versa), and to reveal if any overburden or backfill material had been added over the years.

The soil samples were prepared at Argonne National Laboratory and shipped to a commercial laboratory (LFE Environmental Analysis Laboratories) for radiochemical (fluorometric) and gamma-spectral analyses. The analysis procedures are described in Appendix 4. As shown in Figure 12, sample preparation consisted of weighing the samples and then drying them for about 24 hours at 80°C. All samples were then reweighed, placed into mill jars (8.7 ℓ), and milled until a sufficient amount of the soil sample would pass a No. 30 standard (600 micron mesh) stainless steel sieve. At no point were the rocks and solid material ground or pulverized, since this material would act as a diluent and, hence, lower the reported concentration of deposited radioactive material. The rocks/dross and the sieved material were segregated, bagged, and weighed separately (weights are given in Table 4).

Aliquots of the sieved material were placed in screwtop plastic containers. The amount placed in the containers varied according to the type of analysis to be performed--100 g for gamma-spectral and radiochemical (fluorometric) analysis and 10 g for radiochemical (fluorometric) only.

Every effort was made throughout the sample preparation operations to eliminate cross-contamination. Soil samples suspected of containing elevated amounts of radioactivity were processed in separate equipment from that used to process the soil samples considered to contain background levels. Additionally, all items of equipment were thoroughly scrubbed and air dried before introduction of the next sample.

In addition to the 10 environmental soil samples, two gravel samples (4-S1 and 4-S2) were taken from the roof of Building 55 and three chemical samples (4-S3 through 5) related to the current process were provided by Chemicals Group, Olin Corporation. The preparation of these samples was similar to that described above for the soil samples. However, due to the relatively small mass of the gravel samples, only 25 g and 10 g of samples 4-S1 and 2, respectively, were sent for analysis. Analyses of the gravel and chemical samples included determination of ^{226}Ra by radon emanation, in addition to radiochemical (fluorometric) and gamma-spectral analyses.

SURVEY RESULTS

General

The results of the radiological survey are discussed in this section. The PAC-4G-3 instrument readings and smear results have been normalized to units of dis/min-100 cm^2 using the factors derived in Appendix 2 and are equated to normal uranium, unless otherwise stated. The PAC-4G-3 readings and smear data are reported in net count rates, i.e., the background count rates have been subtracted from the gross count rates prior to conversion to dis/min-100 cm^2. Any alpha contributions have been subtracted from the readings taken in the beta mode so that the corrected values reflect only the beta-gamma readings. The GM exposure rates given in Table 1 include the instrument background of 0.02-0.03 mR/h.

Background levels varied somewhat, due in part to differences in the construction materials used. The average background readings for all modes of operation of the instruments used are given in Appendix 1.

The percentages of surface areas accessible for survey varied from area to area and are indicated in Table 1. The average percentage of the total area that was accessible was 95% for the floors and 90% for the walls.

Instrument and Smear Surveys

Elevated levels of radioactivity, as indicated by measurements clearly above background levels, was found at 47 spots or localized areas in Building 55 or on the roof. In addition, 11 general areas in the building and on the roof exhibited levels slightly above the determined background readings. (See Table 1 and Figures 1 through 9 for the maximum instrument readings at these locations.) Contamination was found mainly on the concrete floors, the overhead beams, and in mixing vats, Kelly filtration devices, and processing tanks. Although the contamination may be due to past AEC activities, it also may be due to the present use of the building involving chemical processing of materials formed from Florida phosphate rock. Gamma-spectral analysis of a yellow residue from a crossover pipe to filtered liquor tank-5 (shown in Fig. 13) indicated

that the contaminant was predominantly normal uranium.* A sample was also taken from the top of the nitric acid tank-1 at Location 107. Gamma-spectral analysis again indicated the contaminant to be normal uranium (see Fig. 14). Because of these findings, all the contamination within the building is equated in this report to normal uranium. Contamination was also found on the roof of the building. Gamma-spectral analysis of a sample of the gravel roof indicated the presence of radium (^{226}Ra) (see Fig. 15 and Table 5); hence, the contamination on the roof has been equated to ^{226}Ra.

Much of the floor area of the main room of the building (see Fig. 1) was found to be contaminated, especially toward the northern side of the room. All of the concrete floor in grids 1 through 11 (see Fig. 1a) and about 30% of the concrete floor in grids 20 and 21 were involved, representing about 600 m^2 total area. Some first level overhead beams (or second level floor beams) were found contaminated, as were some of the sampled mixing vats, Kellys, and processing tanks. The PAC beta-gamma contamination levels ranged from 3.4×10^2 to 1.4×10^6 dis/min-100 cm^2. The maximum beta-gamma reading, 1.4×10^6 dis/min-100 cm^2, was at location 133 on a Kelly filtration device. The PAC alpha contamination levels ranged from background to 5.8×10^3 dis/min-100 cm^2. In 32 of the areas, the GM End-Window exposure rate readings at contact ranged from just barely distinguishable above background to 7 mR/h. The highest GM contact exposure rate reading of 7 mR/h was at location 34 on a steel pump valve flange and at location 102 on top of nitric acid tank-1. Seven GM exposure rate readings taken at 1 m (3 ft) were distinguishable from background. The highest, 0.2 mR/h, was at Location 41, above the concrete floor in grid 20.

The PAC beta-gamma contamination levels on the roof ranged from 1.8×10^3 to 3.4×10^4 dis/min-100 cm^2, equated to ^{226}Ra. The PAC alpha contamination levels on the roof were background. One GM contact exposure rate reading on the roof was just distinguishable from background (i.e., 0.06 mR/h), and none taken at 1 m were distinguishable from the instrument background of 0.02 to 0.03 mR/h.

*The term "normal uranium" refers to uranium which has been separated from its radioactive decay products and other impurities, and which has the normal isotopic percent abundance as found in nature. The normal percent abundances are 0.0054% ^{234}U, 0.720% ^{235}U, and 99.275% ^{238}U (Ref. 6). The less precise definition of normal uranium as 0.7% ^{235}U, 99.3% ^{238}U, and a trace of ^{234}U is sometimes used for brevity in discussions. The term "natural uranium" denotes uranium and all decay products as found in its natural state in the earth, and is sometimes incorrectly referred to as normal uranium. Appendix 5 contains the detailed calculation of the specific activity of normal uranium.

Radioactive contamination was detected on 15 smears, indicating the presence of removable contamination. The results are included in Table 1. Beta-gamma levels ranged from background to 850 dis/min-100 cm^2, and alpha levels ranged from background to 640 dis/min-100 cm^2. No contamination statistically greater than the instrument background of the gas-flow proportional counters, as given in Appendix 1, was detected on any other smears. The locations at which all smears were taken are shown in Figures 1 through 9.

Results of the instrument and smear surveys were compared with both the American National Standards Institute (ANSI) Draft Standard N13.12, "Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use," and the NRC's "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licensees for By-Product, Source, or Special Nuclear Material" (see Appendix 6). The limits for uranium were used as the basis of comparison for the contamination levels within the buildings, and the limits for radium were used as the basis of comparison for the contamination levels on the roof.

The allowable limit in the ANSI Draft Standard for uranium activity is 5000 dis/min-100 cm^2 total, of which only 1000 dis/min-100 cm^2 can be removable. These levels may be averaged over 1.0 m^2, provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. The NRC Guidelines for uranium are stated as follows: the average is 5000 dis/min-100 cm^2 alpha, the maximum is 15,000 dis/min-100 cm^2 alpha, and the removable is 1000 dis/min-100 cm^2 alpha. The measurements used for the average may not be averaged over more than 1 m^2, and the maximum level applies to an area of not more than 100 cm^2. Also, 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 mg/cm^2 of total absorber. The ANSI Draft Standard is identical to the NRC Guidelines for uranium; however, the ANSI limits do not exclude the determination of uranium by beta-gamma activity, whereas the NRC Guidelines are stated in terms of alpha activity only.

The allowable limit in the ANSI Draft Standard for radium is 20 dis/min-100 cm^2 removable, and the limits are such that the total (fixed plus removable) activity must be nondetectable using instruments calibrated to measured at least 100 pCi for the contaminant uniformly spread over 100 cm^2. The NRC Guidelines

for radium are stated as follows: the average is 100 dis/min-100 cm^2, the maximum is 300 dis/min-100 cm^2, and the removable is 20 dis/min-100 cm^2. The measurements used for the average may not be averaged over more than 1 m^2, and the maximum level applies to an area of not more than 100 cm^2.

Thirty-three spots or localized areas and three larger general areas within Building 55 exceeded the acceptable surface contamination limits for uranium, and the general roof area exceeded the acceptable limits for radium. Additionally, two spots or localized areas on the roof substantially exceeded those limits. The locations in which contamination was found to be greater than the acceptable limits are listed in Table 2.

Air Samples

Results of the analyses of air samples collected at selected locations in the building are presented in Table 3. Techniques detailed in Appendix 3 were used to determine the radon-222 concentration and decay product Working Levels (WL) at each location. The results ranged from 0.0014 to 0.0061 WL and were within the range of values normally expected for background concentrations. Under the U.S. Surgeon General's Guidelines in 10 CFR 712 (see Appendix 6), concentrations of radon daughters of less than 0.01 WL above background do not indicate a need for remedial action. Radon concentrations, as determined from the radon decay-product measurements, ranged from 0.14 to 0.61 pCi/ℓ, well below the concentration guide of 3 pCi/ℓ for an uncontrolled area, as given in the Department of Energy's "Requirements for Radiation Protection." No long-lived radionuclides were detected on any air sample.

Soil and Material Samples

Results of the gamma-spectral, uranium-fluorometric, and other analyses performed on the soil and material samples by LFE Environmental Analysis Laboratories are listed in Table 5. The analyses of the ten soil samples collected on the grounds of Chemicals Group, Olin Corporation, indicated uranium concentrations ranging from 0±0.2 to 87.7±4.8 pCi/g and radium decay-chain concentrations ranging from 0.18±0.03 to 18.1±0.91 pCi/g. As indicated in Table 6, levels of natural uranium in background samples collected in the Chicago area ranged from 0.6±0.1 to 2.2±0.3 pCi/g. Eight of the 10 soil samples taken on the grounds of the site had uranium concentrations in excess of the 2.2 pCi/g back-

ground level. Two samples, 4-S8 and 4-S9, collected just south and east of Building 55, respectively, contained significantly elevated levels of uranium (16±1 to 87.7±4.8 pCi/g) in all four segments of each sample. There currently are no standards specifying limiting concentrations for uranium in soil, but the measured uranium concentrations in the top segment of each of the two samples (87.7±4.8 and 53±3 pCi/g, respectively) contained uranium concentrations in excess of the proposed interim soil limit of 40 pCi/g for decommissioning and decontamination projects (Ref. 2 and Appendix 6). The two samples also contained elevated levels of radium (7.96±0.40 and 18.1±0.91 pCi/g, respectively) in excess of the 5 pCi/g limit as specified in the Environmental Protection Agency's 40 CFR 192, Subpart B ("Standards for Cleanup of Open Lands and Buildings Contaminated with Residual Radioactive Materials from Inactive Uranium Processing Sites") (see Appendix 6). The elevated levels were judged by the survey team to be the result of the release of some uranium-containing residue from the plant.

The analyses also indicated concentrations of ^{137}Cs ranging from $\leq$ 0.02 to 2.02±0.08 pCi/g and concentrations of the thorium decay-chain products ranging from 0.27±0.06 to 1.51±0.20 pCi/g. Analyses of the background samples collected in the Chicago area indicated concentrations of ^{137}Cs ranging from 0.8±0.3 to 3.0±0.7 pCi/g and of thorium ranging from 0.18±0.02 to 0.60± 0.04 pCi/g. Hence, the ^{137}Cs and thorium decay chain concentrations in soil samples collected around Building 55 were essentially within the same range as natural background concentrations.

The results of the analyses of the two roof gravel samples (4-S1 and 4-S2) and three chemical samples (4-S3 through 5) are included in Table 5. The analyses of the gravel samples indicated concentrations of the radium decay chain of 14.6±0.7 and 15.1±0.8 pCi/g and concentrations of uranium of 136±7 and 46±2 pCi/g, all in excess of either the EPA limit for radium or the proposed interim limit for uranium, as applicable. The analyses of the ground phosphate rock indicated concentrations of the radium decay chain of 36.0±1.8 pCi/g and uranium of 79.6±8.2 pCi/g, each in excess of the respective limits. These results are consistent with typical natural uranium concentrations of Florida phosphate rock. The analyses of the disodium phosphate indicated concentrations of the radium decay chain of 0.7±0.3 and 0.64±0.15 pCi/g and concentrations of uranium of 87±4 and 184±7 pCi/g, consistent with typical disequilibrium concentrations following dissolution by sulfuric acid (Ref. 1).

ESTIMATED EXTENT OF CONTAMINATION

Any estimate of the total mass and volume of radioactively contaminated material that would be generated by remedial action at this facility is subject to many uncertainties. For example, one can only surmise as to the actual depth of contamination within concrete and steel and the depth of contamination involved at soil sample locations 4-S8 and 4-S9. In the case of this particular survey, which was performed prior to the establishment of the requirement that mass and volume of contaminated materials be estimated, only limited data are available. Therefore, presumably conservative assumptions have been made on the basis of professional judgment.

To estimate the volume of soil that would be removed in any remedial action, it has been assumed that excavation of a 2 m^2 area to a depth of 1 m would be sufficient to include all significant contamination. For the case of the contaminated concrete floor and steel beams and tanks, two alternatives are possible. If decontamination proved to be feasible, the concrete and steel would not become radioactive waste, but the decontamination residues would. This is tabulated as Option A in Table 7. If decontamination was not feasible or proved to be unsuccessful, the concrete and steel would need to be treated as radioactive waste as indicated by Option B in Table 7. In this option, it has been assumed that contamination on concrete would require removal to a depth of 5 cm (2 in); contamination on steel would require removal of its entire thickness. In Option A, the mass of radioactive waste generated during decontamination of the concrete and steel was arbitrarily taken as 1% of the mass of the material being decontaminated. This value would be strongly influenced by the method of decontamination employed, assuming a successful method could be found.

Estimates of the total activity of contaminated material are likewise subject to some uncertainties because of survey limitations. Unless otherwise stated, all readings of dis/min-100 cm^2 (as reported in Table 2) are equated to thin flat-plate standards. No corrections are made for absorption by surface media since any correction factors would, in themselves, only be rough estimates. Hence, estimates of activity in surface media could be somewhat underestimated.

Despite these uncertainties and limitations, and based on the assumptions above, estimates of volume, mass, and activity have been made for the several types of materials present and are presented in Table 7. As indicated in the table, Option A would generate 24 m^3 of material with a mass of 46,000 kg, while

Option B would generate 26 m^3 of material with a mass of 54,000 kg. The activity of the material would be about the same for either option, an estimated 200 μCi as normal uranium and 61 μCi as radium-226.

DOSE AND POTENTIAL HAZARD EVALUATION

The survey data on surface contamination, external penetrating radiation, radioactivity on airborne particulates, and radioactivity in soil and material samples at the Chemicals Group, Olin Corporation may be evaluated in terms of the dose equivalent commitments that potentially exposed persons could receive. These doses can then be compared to the appropriate standards and/or natural background radiation doses or used to estimate risks of health effects.

The appropriate radiation protection standards for external and internal exposure of individuals and population groups in uncontrolled areas are given in the Department of Energy's publication "Requirements for Radiation Protection" (see Appendix 6) and are expressed as the permissible dose or dose commitment annually (in mrem) beyond that received from background radiation and medical exposures.

Natural background radiation doses consist of an external penetrating dose from cosmic and terrestrial sources and an internal dose from the inhalation/ingestion of radioactivity from cosmogenic and terrestrial sources. The average annual natural background doses for the U.S. population are 54 mrem external and 28 mrem internal to the whole body, 54 mrem external and 125 mrem internal to the lung, and 54 mrem external and 117 mrem internal to the bone (osteocytes) (Ref. 7). The total whole body, lung, and bone doses are thus 82 mrem, 179 mrem, and 171 mrem per year, respectively. Background radiation is discussed in more detail in Appendix 8.

Estimates of radiological risks resulting from specific doses are usually based on risk factors as provided in reports by the International Commission on Radiological Protection (ICRP) (Ref. 8), National Research Council Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR) (Refs. 9, 10), or United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) (Ref. 11). By multiplying the estimated dose equivalent by the appropriate risk factor, one can obtain an estimate of the risk or probability of the occurrence of health effects such as cancers and hereditary effects to an individual or his descendants as a result of that exposure. The evaluation of

risk factors is presently subject to large uncertainties and continual revision, and is the subject of considerable controversy. For these reasons, such risks have not been calculated for this report.

Potential doses resulting from exposure to radioactivity were calculated for a pathway or scenario that could result in the presumed maximum external penetrating radiation dose and for a pathway that could result in the presumed maximum internal radiation dose from inhalation/ingestion of radioactive material. Since the analyses of radioactivity on airborne particulates indicated natural background levels only, no pathways were considered for these sources; only surface contamination and external penetrating radiation were considered. Details of these dose calculations are discussed in Appendix 7.

The maximum potential external penetrating radiation dose from contamination was calculated to be 340 mrem per year. This represents an increase of about 410% over the 82-mrem natural background whole-body dose and is 68% of the 500-mrem standard for an individual in an uncontrolled area.

The presumed internal radiation dose commitments from potential inhalation/ingestion of contamination were calculated to be 8300 mrem to the lung, 120 mrem to the bone, 26 mrem to the kidney, and 250 mrem to the whole-body. These are 50-year dose commitments and represent the total dose equivalent that would be accumulated in the body or specific critical organs over a 50-year period from inhalation/ingestion in the first year. Fifty-year dose commitments are always as large or larger than first-year annual doses; hence, all comparisons to annual dose standards are of a conservative nature. For the lung, bone, and kidney, these doses represent additions of 4600%, 70%, and 32% to the 179-mrem, 171-mrem, and 82-mrem annual natural background lung, bone, and kidney doses, respectively, and 550%, 8%, and 1.7% of the 1500-mrem annual standard for an individual in an uncontrolled area. For the whole body, the calculated dose represents an increase of 300% to the 82-mrem annual natural background whole-body dose and 50% of the 500-mrem annual standard for an individual in an uncontrolled area. Few individuals are expected to acquire such doses or dose commitments annually.

In order to reduce the potential for radiation exposure, remedial measures such as stabilization of the contamination in place would be applicable as a short-term measure. To reduce the risk in the event that building modifications take place in the future, health physics procedures and coverage are recommended. The long-term solution would involve decontamination by removal of the radioactive residues from the areas in the facility where contamination was detected.

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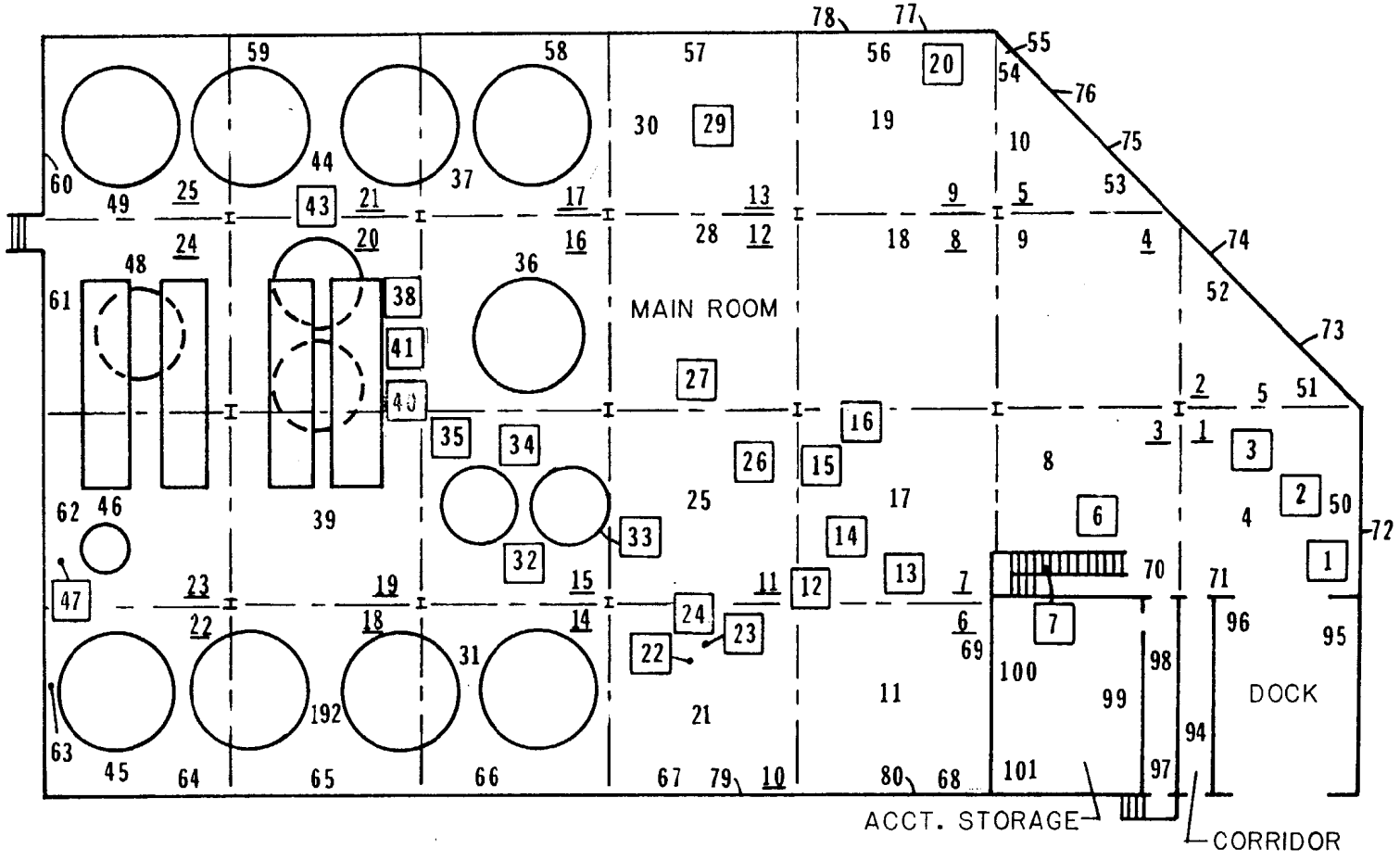
FIGURE 1a

FLOOR AND WALL SURVEY LOCATIONS ON FIRST LEVEL

ANL-HP DWG. NO. 79-40



0 10 METERS

N

n GRID NUMBER

n SMEAR

[n] DIRECT AND/OR SMEAR READING ABOVE BACKGROUND

I SUPPORT COLUMN (DEFINES GRID CORNERS)

FIGURE 1b
AIR SAMPLE AND OVERHEAD SURVEY LOCATIONS ON FIRST LEVEL
ANL-HP DWG. NO. 79-40a

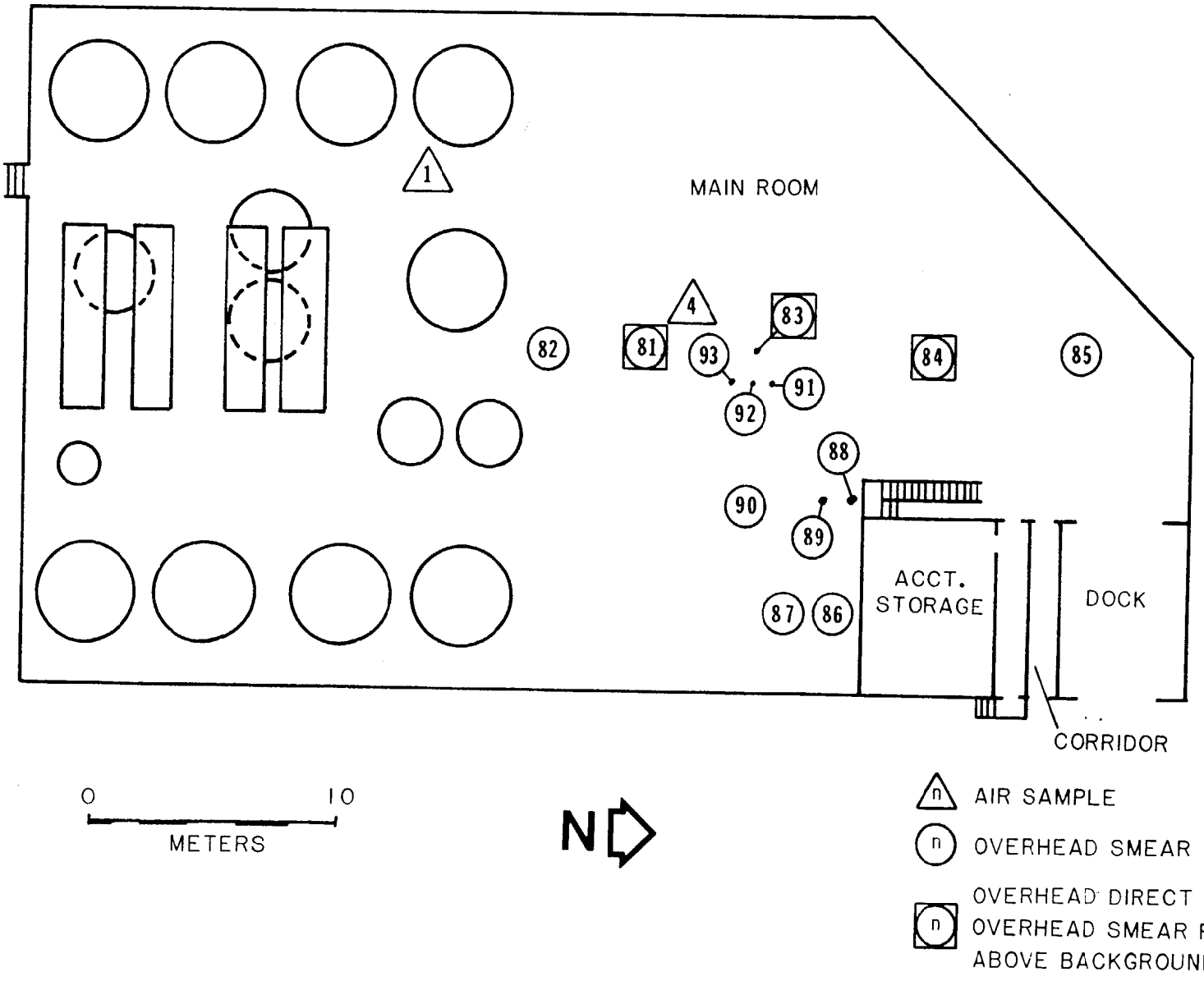


FIGURE 2

AIR SAMPLE AND SURVEY LOCATIONS ON SECOND LEVEL

ANL-HP DWG. NO. 79-41

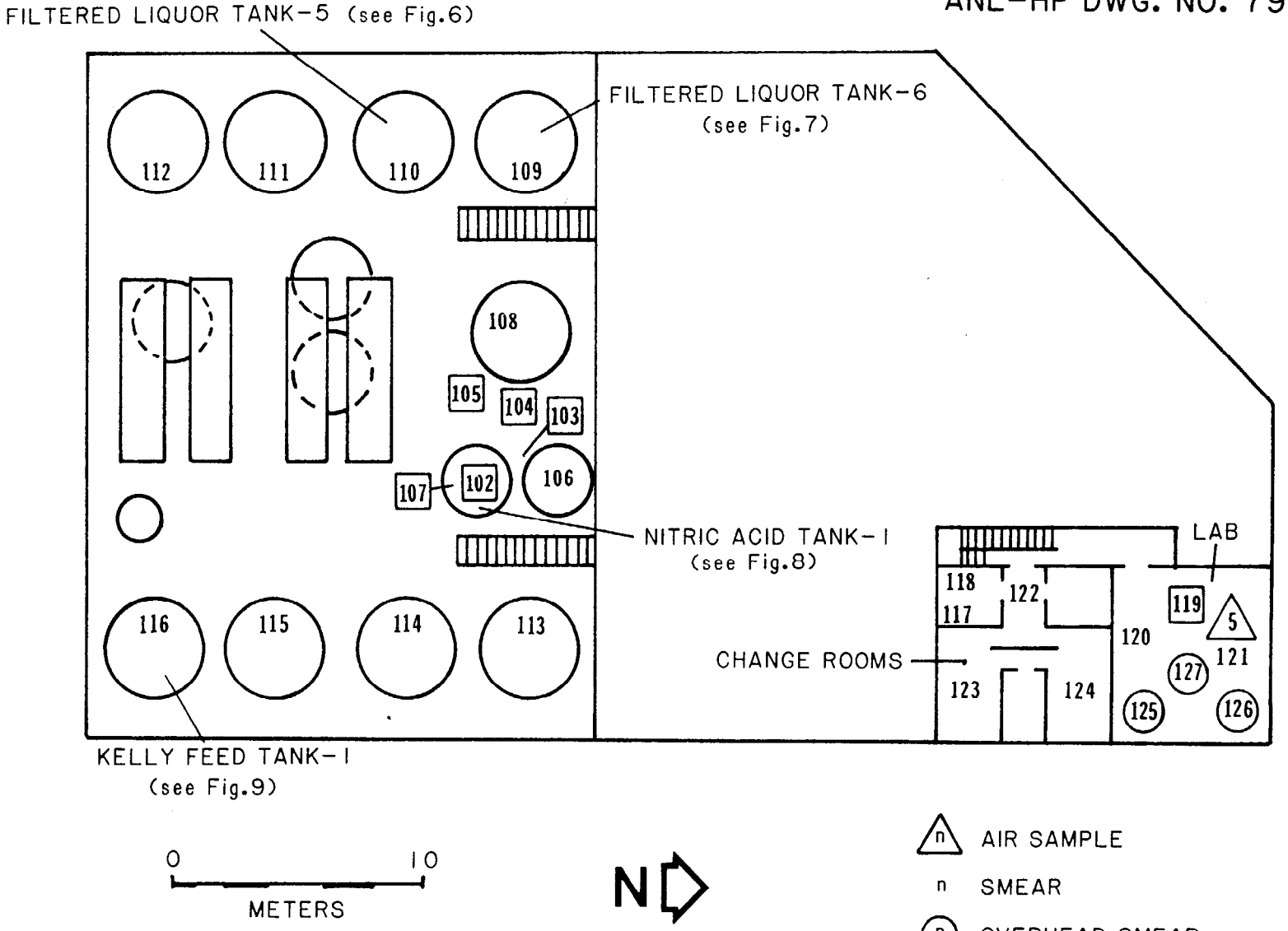


FIGURE 3

AIR SAMPLE AND SURVEY LOCATIONS ON THIRD LEVEL

ANL-HP DWG. NO. 79-42

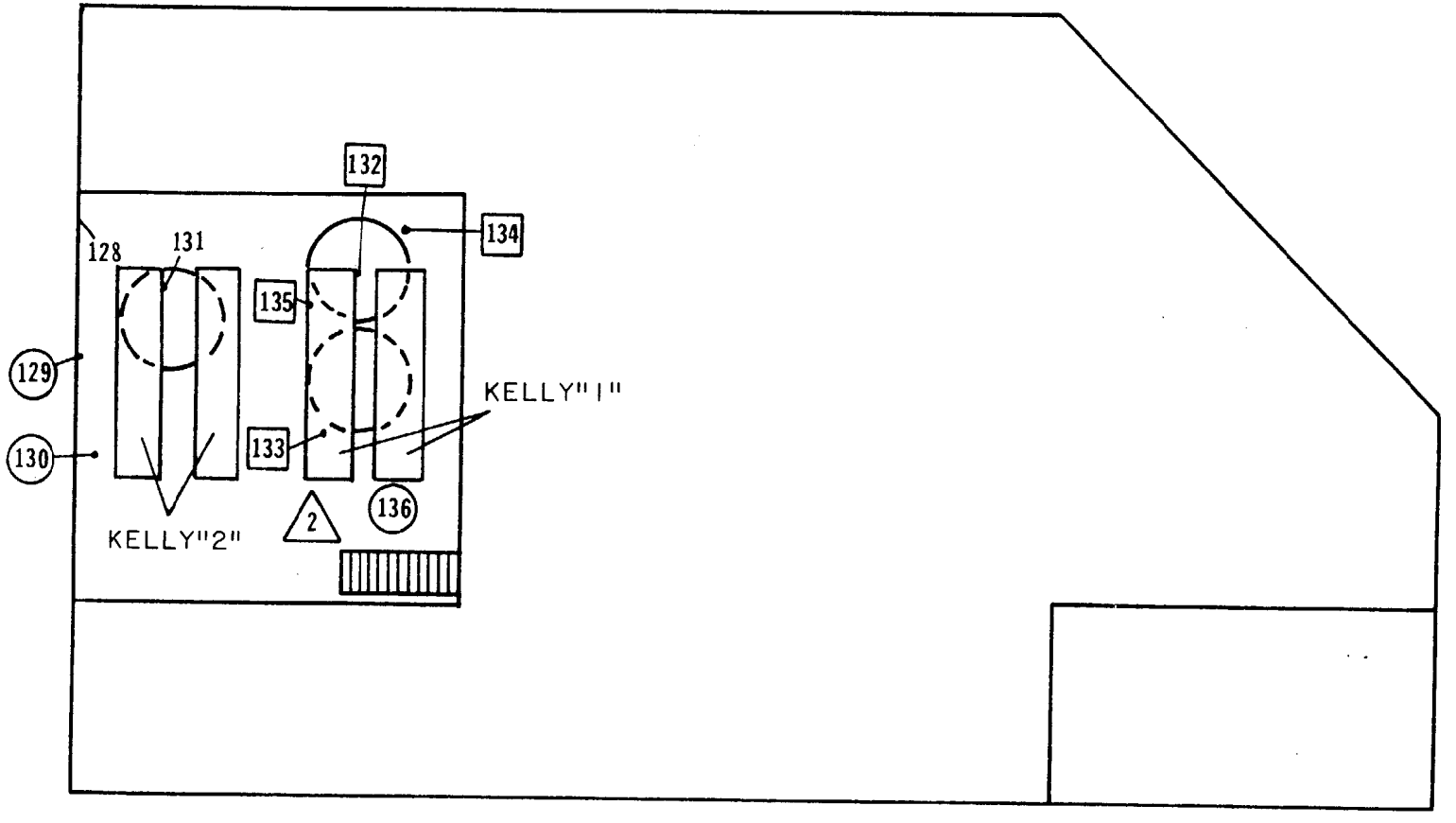


FIGURE 4
SURVEY LOCATIONS ON FOURTH LEVEL

ANL-HP DWG. NO. 79-43

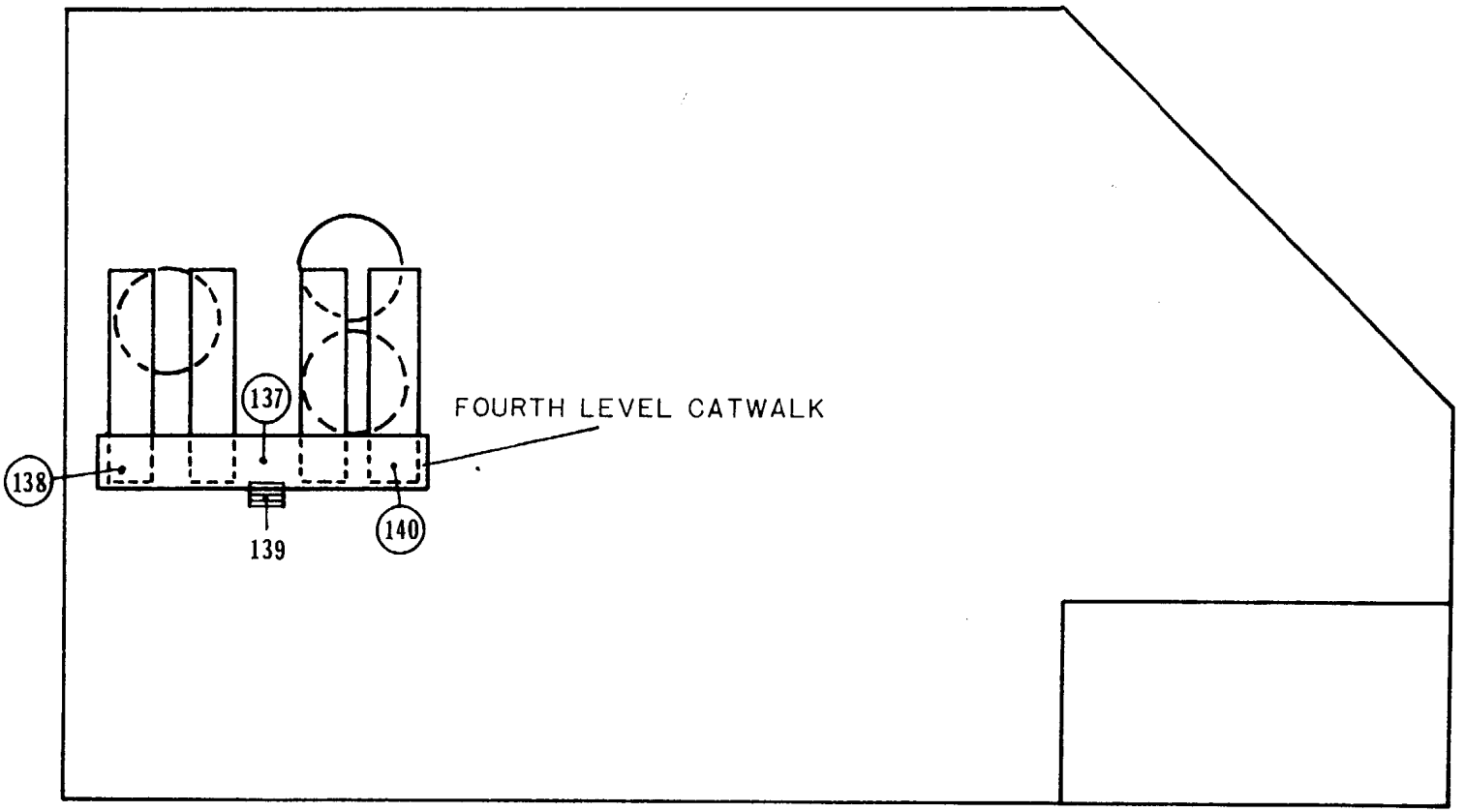


FIGURE 5

AIR SAMPLE AND SURVEY LOCATIONS ON ROOF

ANL-HP DWG. NO. 79-44

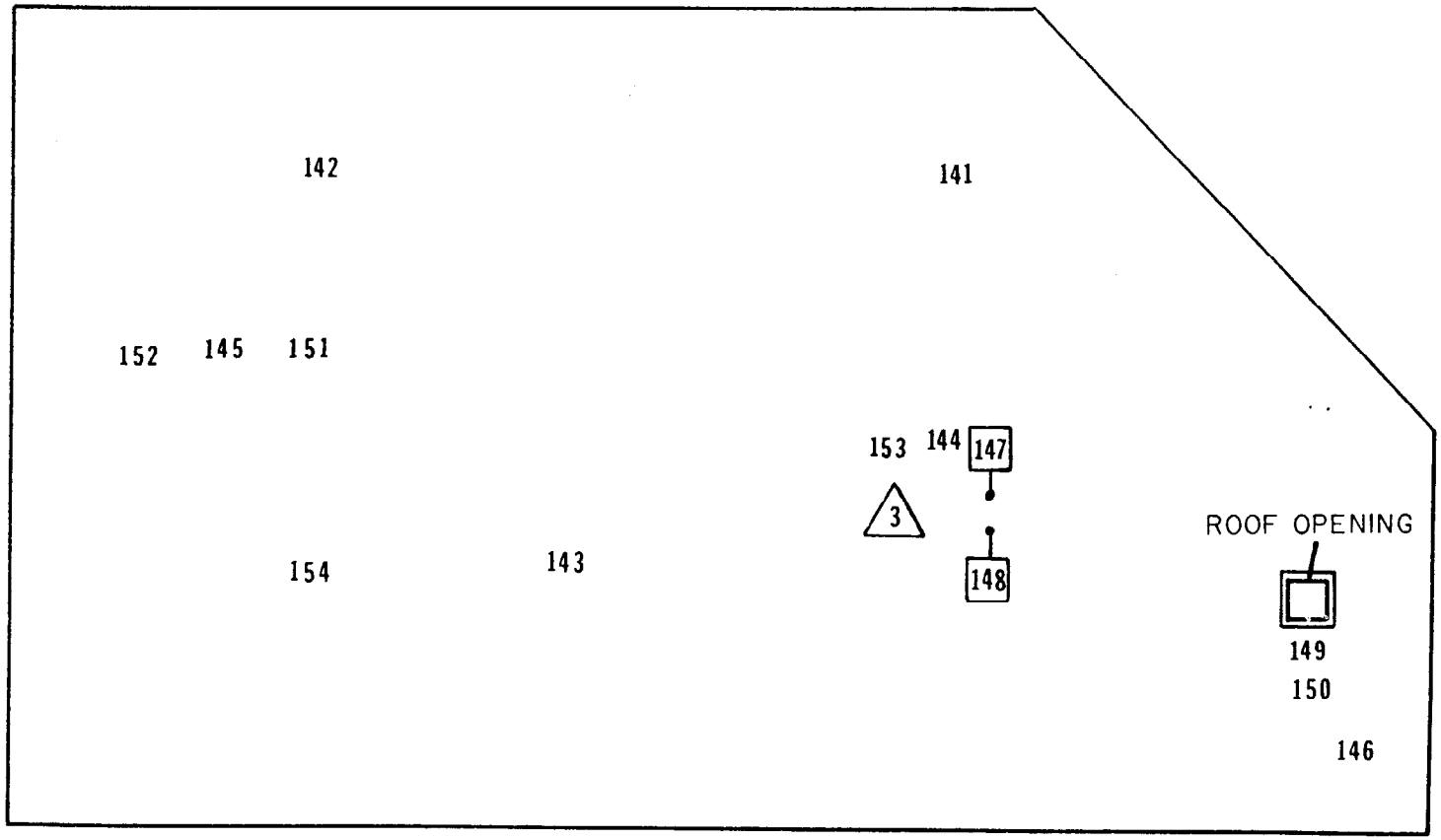
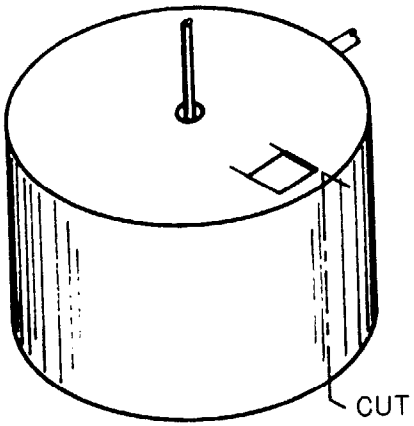
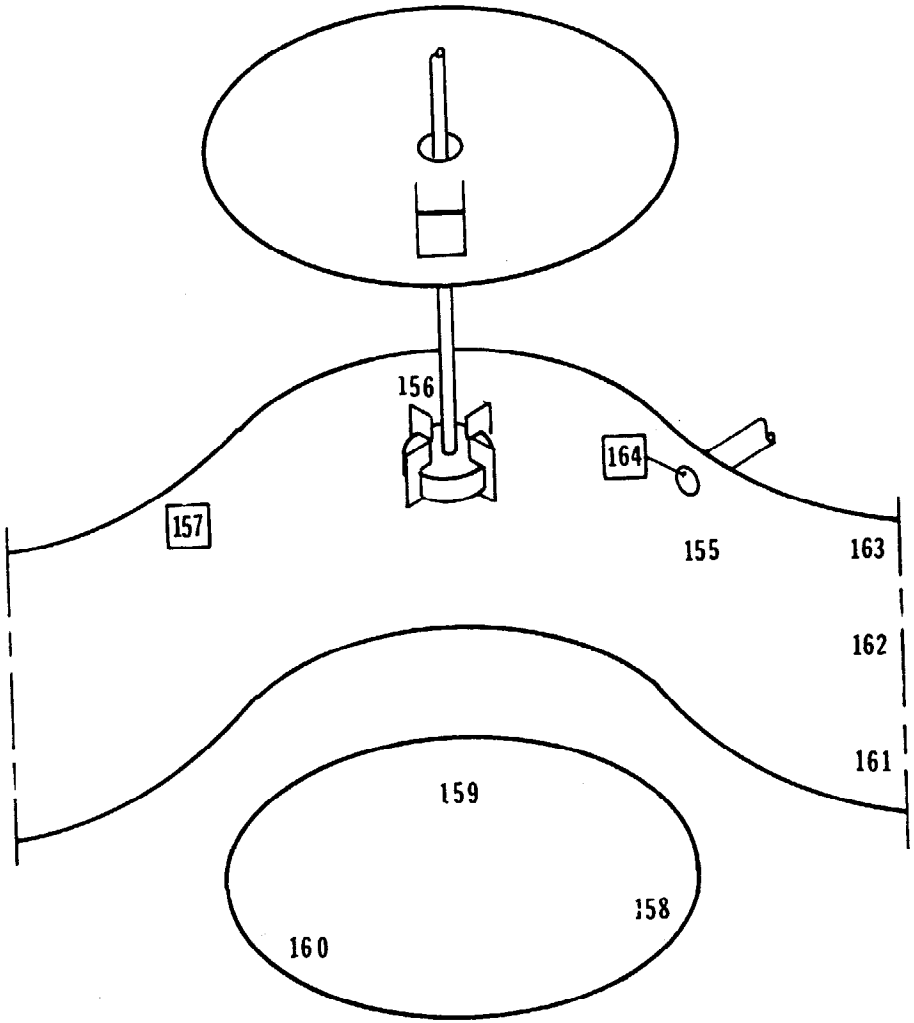


FIGURE 6
SURVEY LOCATIONS IN FILTERED LIQUOR TANK-5

ANL HP DWG. NO. 79-46



10' x 15' DIA.



NO SCALE

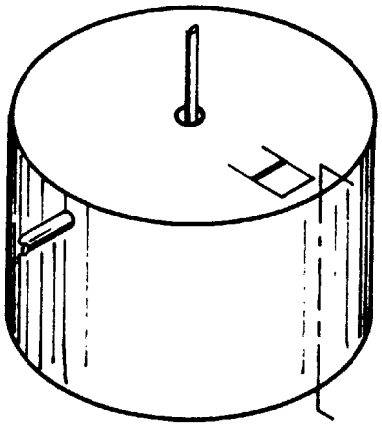
n SMEAR

[n] DIRECT AND/OR SMEAR READING ABOVE BACKGROUND

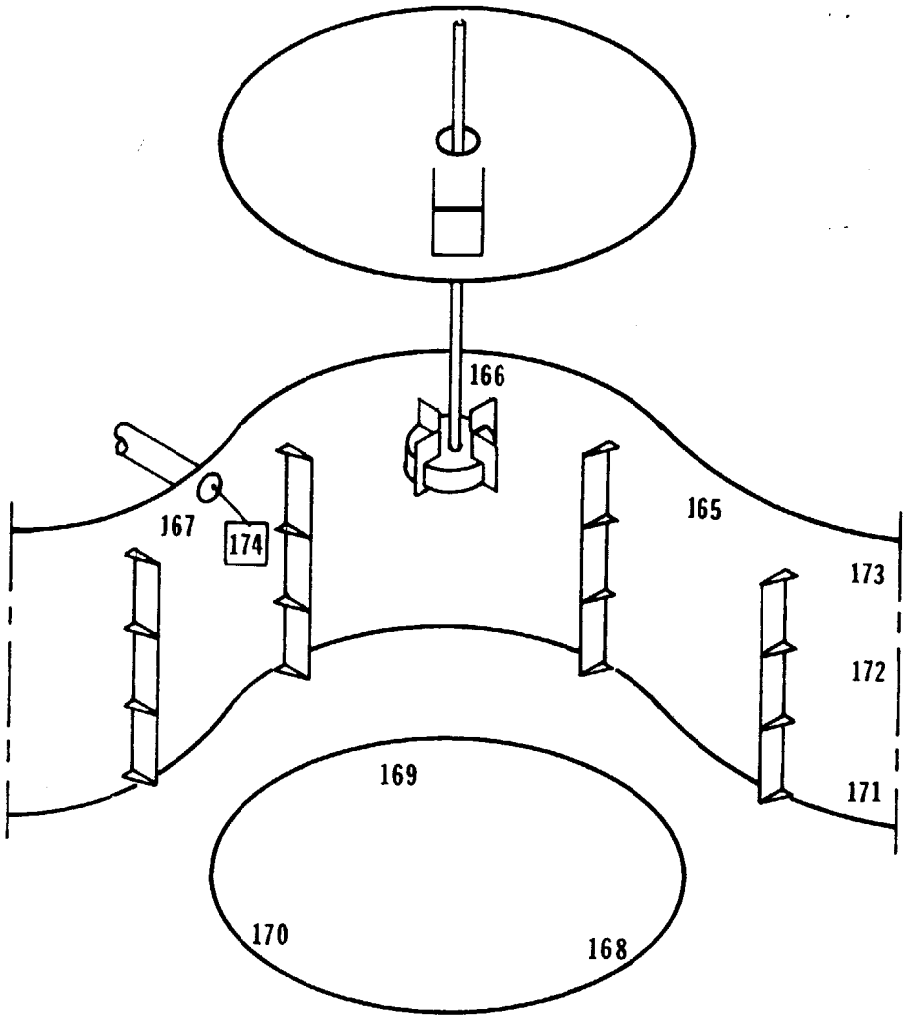
FIGURE 7

SURVEY LOCATIONS IN FILTERED LIQUOR TANK-6

ANL-HP DWG. NO. 79-47



10' x 15' DIA.



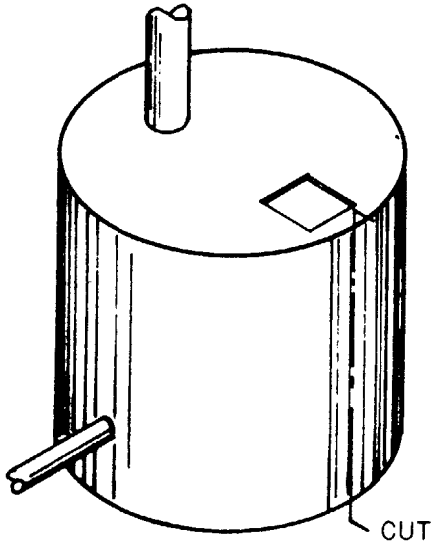
NO SCALE

n SMEAR

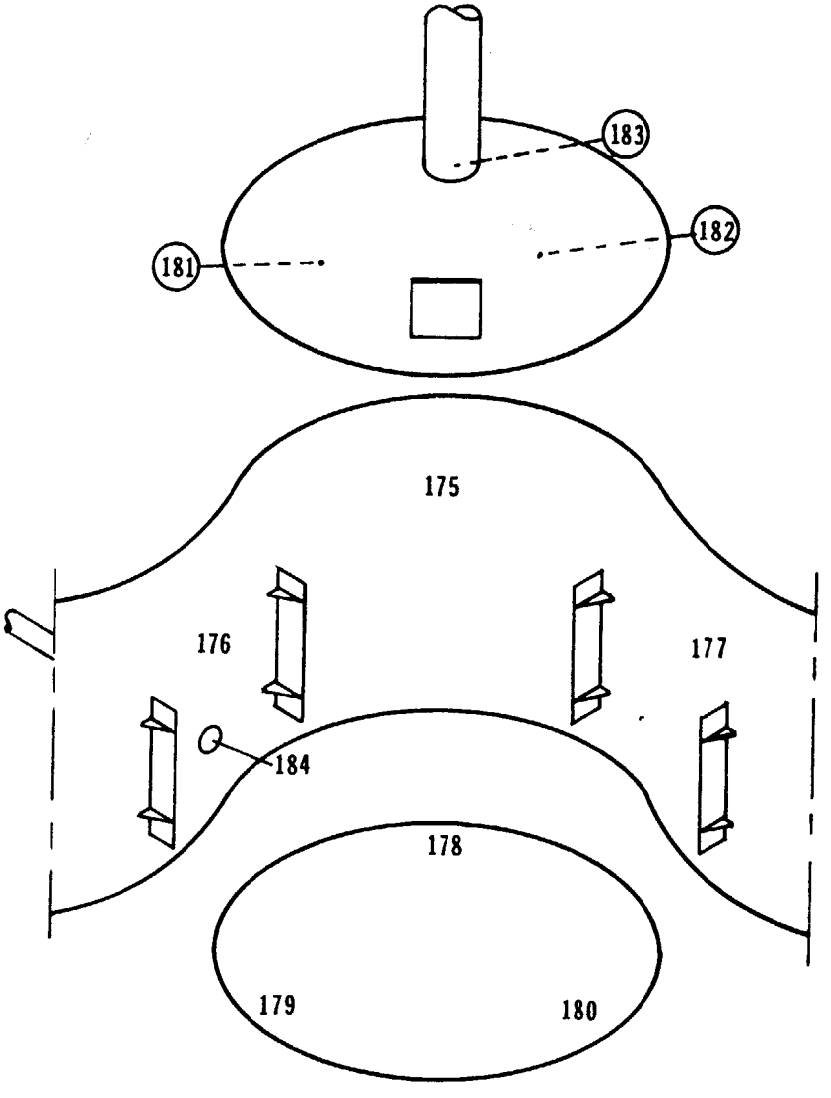
[n] DIRECT AND/OR SMEAR READING ABOVE BACKGROUND

FIGURE 8
SURVEY LOCATIONS IN NITRIC ACID TANK-1

ANL-HP DWG. NO 79-45



10' x 10' DIA.



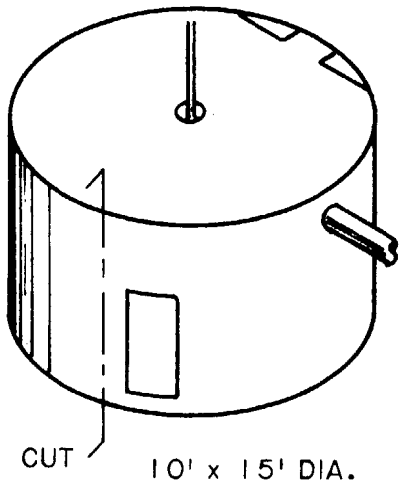
NO SCALE

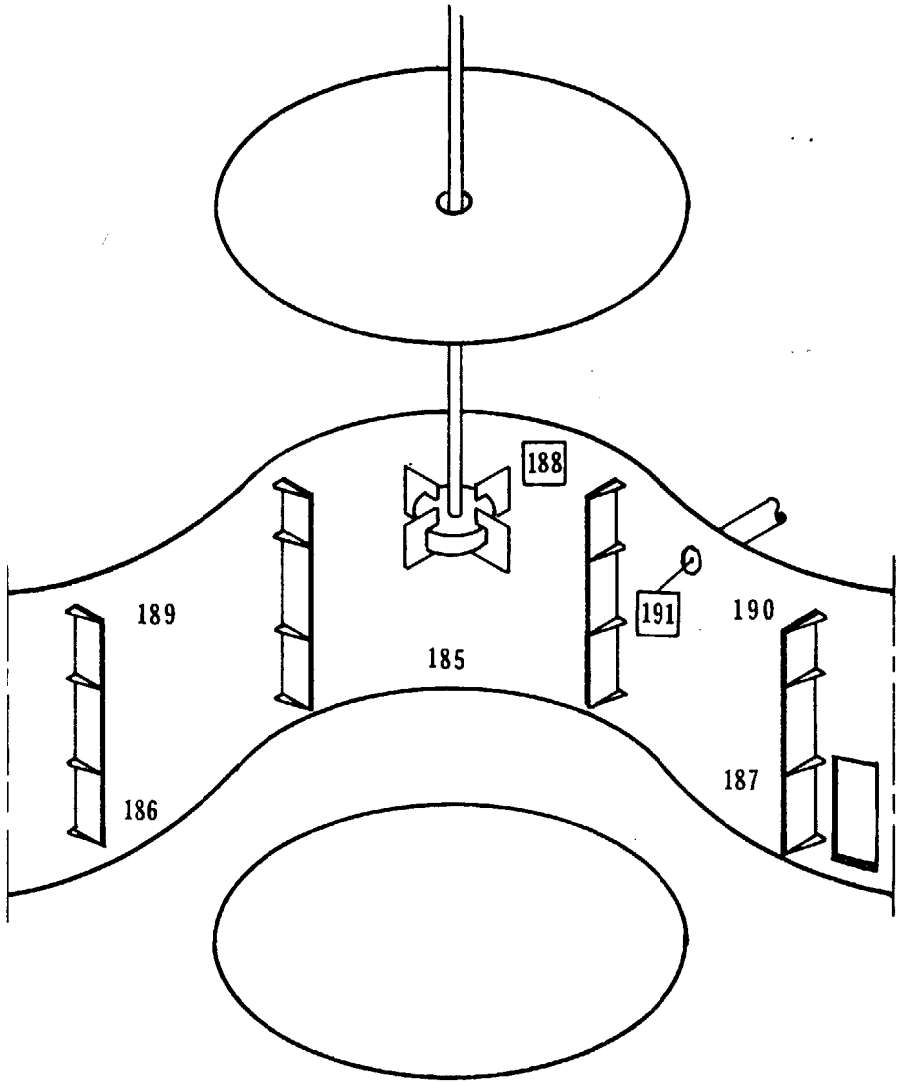
n SMEAR
(n) OVERHEAD SMEAR

FIGURE 9

SURVEY LOCATIONS IN KELLY FEED TANK-1

ANL-HP DWG. NO. 79-48





n SMEAR

[n] DIRECT AND/OR SMEAR READING ABOVE BACKGROUND

NO SCALE

FIGURE 10

ENVIROMENTAL SOIL SAMPLE LOCATIONS

ANL-HP DWG. NO. 79-37

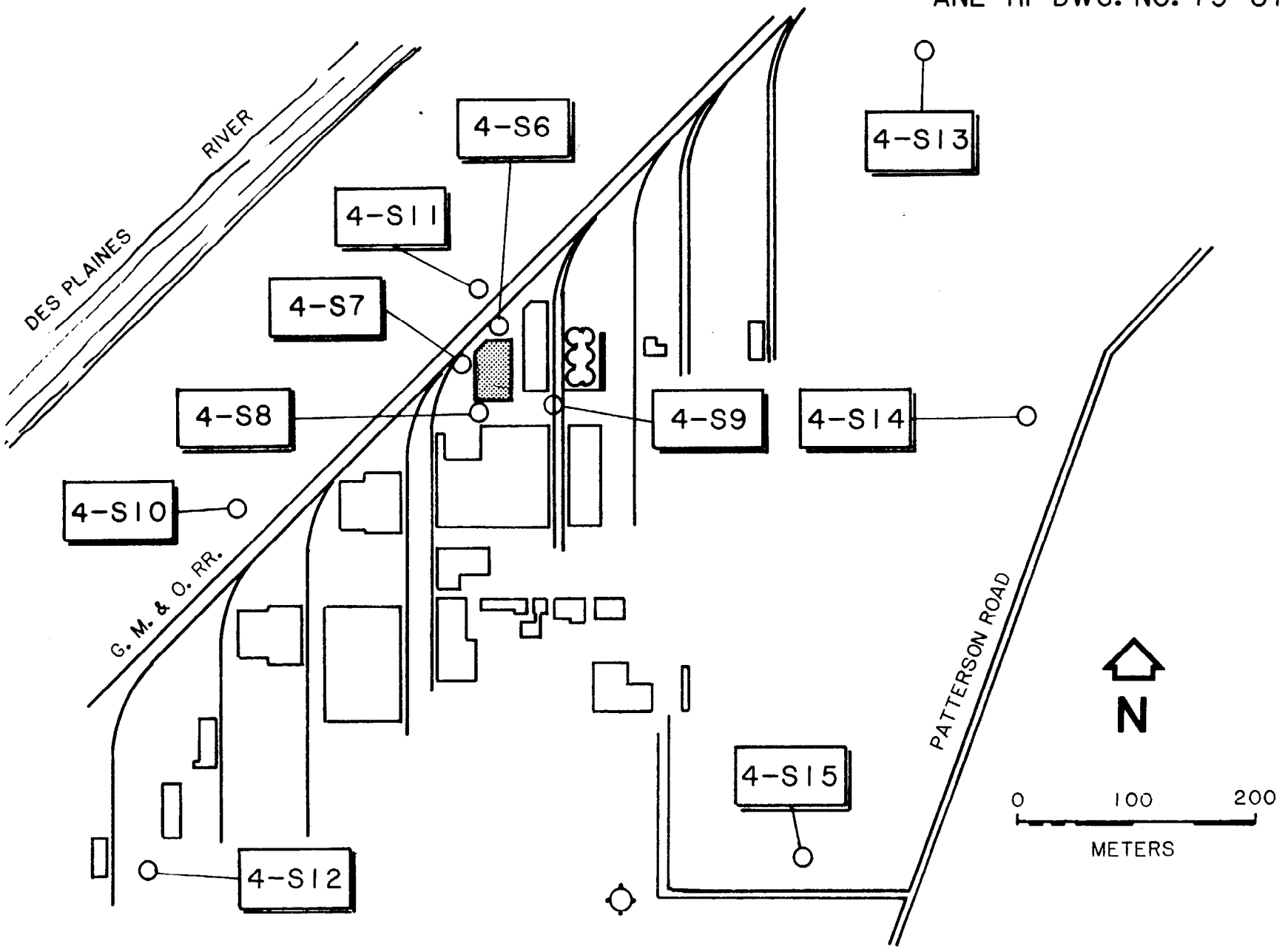


FIGURE 11
OLIN CHEMICAL SITE LOCATION

ANL-HP DWG. NO. 79-38

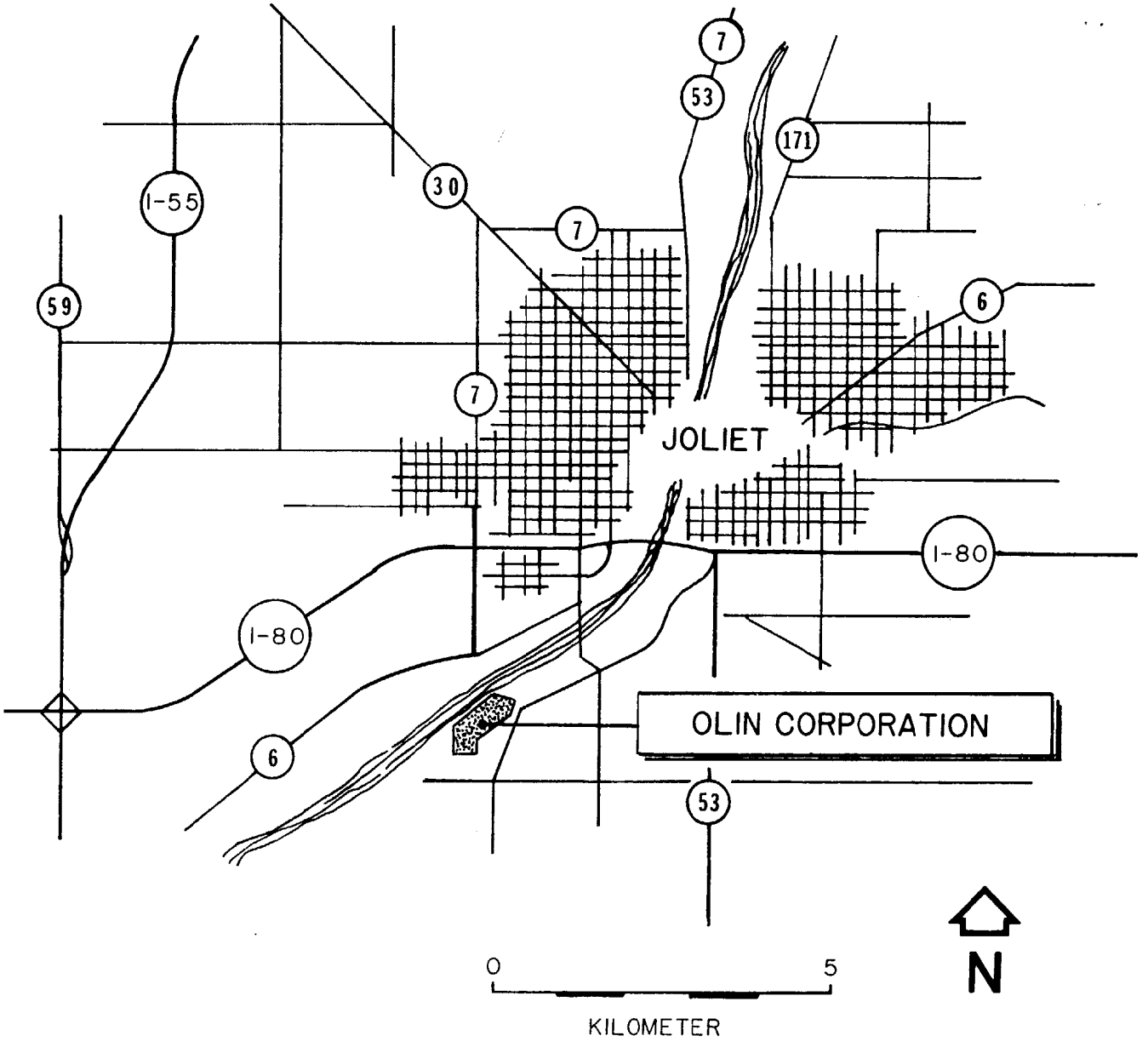


FIGURE 12

SOIL SAMPLING PROCEDURE AND PROCESSING DIAGRAM

ANL-HP DWG. NO. 78-2

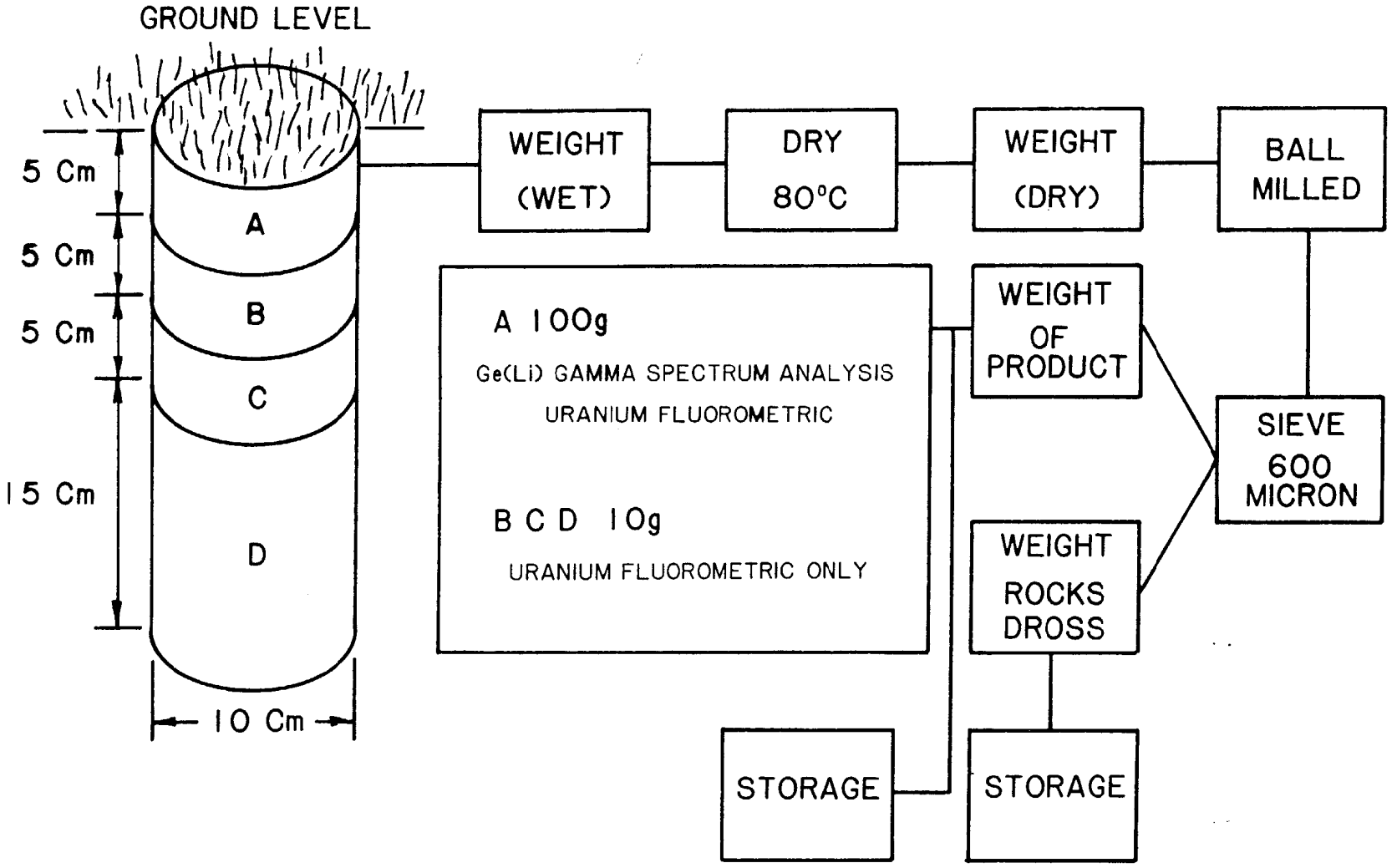


FIGURE 13

GAMMA SPECTRUM OF SAMPLE FROM CROSSOVER PIPE TO LIQUOR TANK-5

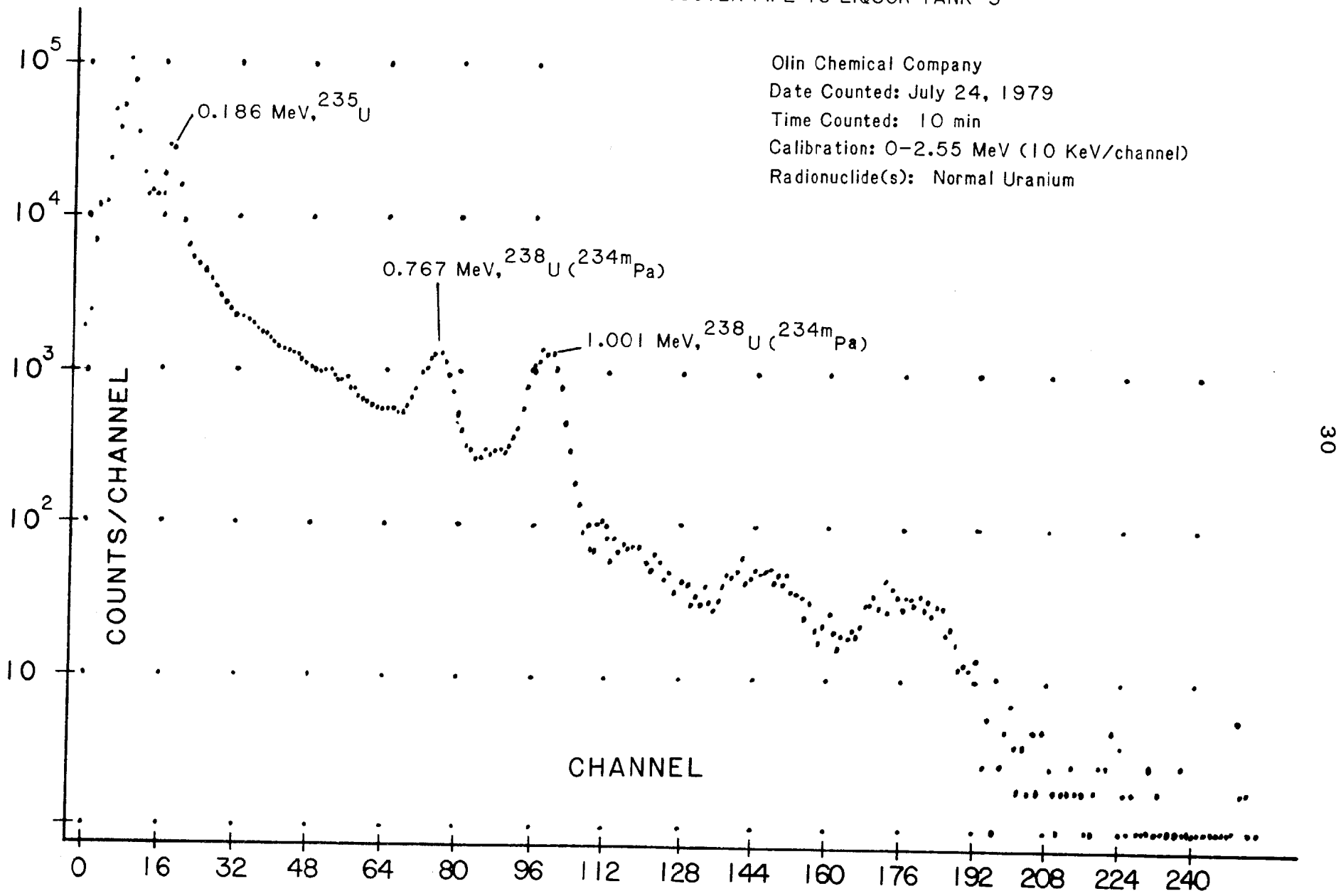


FIGURE 14

GAMMA SPECTRUM OF SAMPLE FROM TOP OF NITRIC ACID TANK – 1

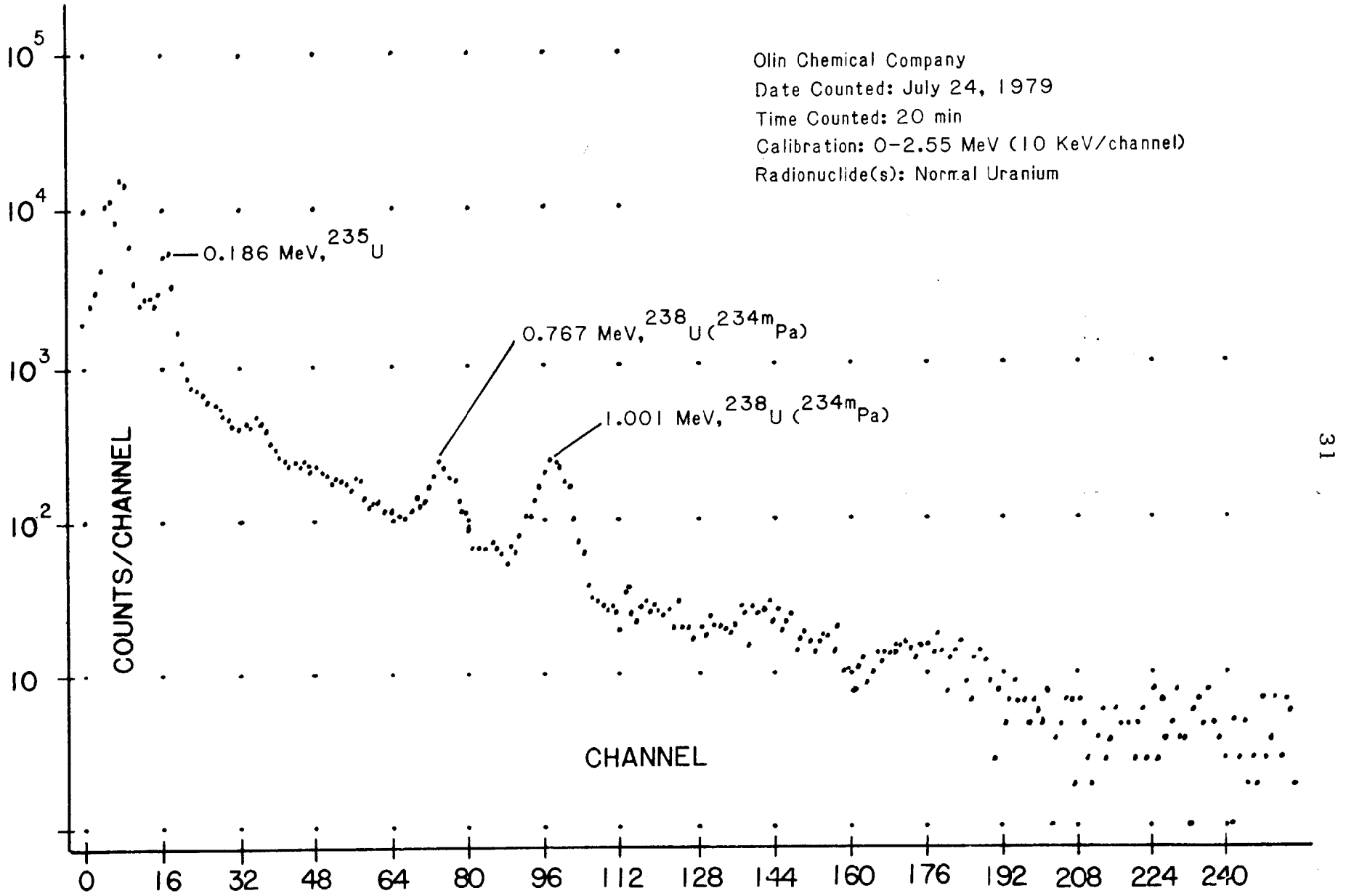


FIGURE 15

GAMMA SPECTRUM OF ROOF GRAVEL

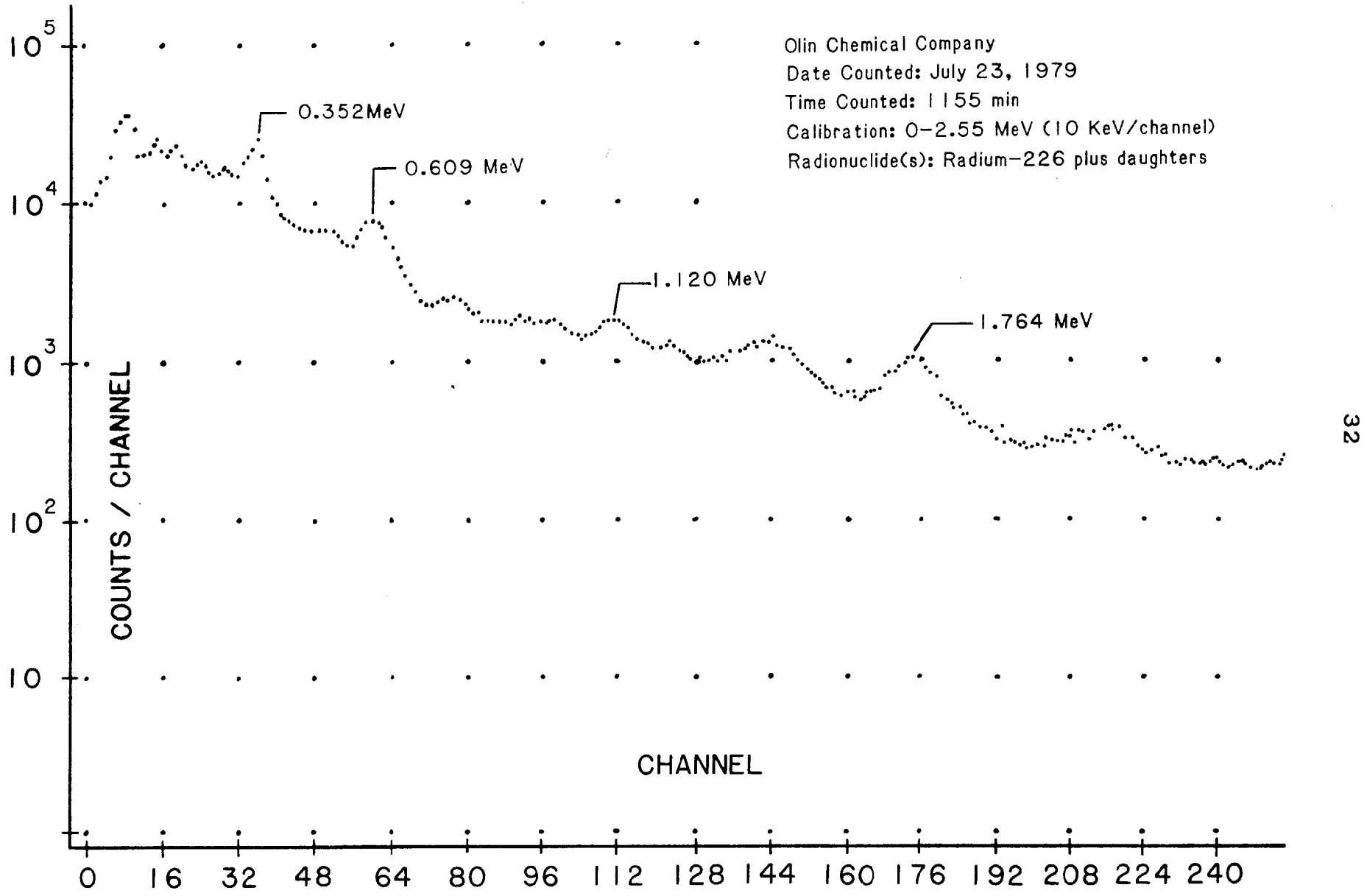


TABLE 1

DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm^2)		End Window (mR/h)		Smear Results (dis/min-100 cm^2)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
1st level of Main Room:	80	70							
Grids 1 through 9			0.0061						Air Sample 4
				1.2×10^4	BKGD[c]	0.1	BKGD	NST[f]	Location 1, spot on concrete floor.
				8.2×10^3	BKGD	0.05	BKGD	NST	Location 2, spot on concrete floor
				6.9×10^2	BKGD	0.04	BKGD	NST	Location 3, spot on concrete floor
				4.3×10^3	BKGD	BKGD	BKGD	BKGD	Location 6, spot on concrete floor
				4.3×10^3	BKGD	BKGD	BKGD	BKGD	Location 7, spot on concrete floor beneath stairs.
				2.7×10^4	BKGD	BKGD	BKGD	BKGD	Location 12, area on concrete floor
				1.1×10^4	BKGD	BKGD	BKGD	BKGD	Location 13, spot on concrete floor
				1.6×10^4	BKGD	BKGD	BKGD	BKGD	Location 14, spot on concrete floor

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm²)		End Window (mR/h)		Smear Results (dis/min-100 cm²)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
				$1.1x10^4$	BKGD[c]	BKGD	BKGD	BKGD	Location 15, spot on concrete floor
				$1.1x10^4$	BKGD	BKGD	BKGD	BKGD	Location 16, Spot on concrete floor
				$1.6x10^4$	BKGD	0.05	BKGD	BKGD	Location 20, spot on concrete floor
				$1.1x10^3$	BKGD	BKGD	BKGD	BKGD	Rest of survey, general contamination on floor
Grids 10 and 11				$2.3x10^4$	$1.2x10^3$	0.1	BKGD	BKGD	Location 22, spot on concrete floor
				$3.4x10^4$	$1.2x10^3$	0.05	BKGD	BKGD	Location 23, spot on concrete floor
				$3.4x10^4$	$1.2x10^3$	0.2	0.05	BKGD	Location 24, spot on concrete floor
				$1.6x10^5$	$2.9x10^3$	0.4	0.04	α =6[e] $\beta\gamma$=18	Location 26, spot on concrete floor
				$2.7x10^3$	BKGD	BKGD	BKGD	BKGD	Rest of survey, general contamination on floor
Grids 12 through 19			0.0026						Air Sample 1
				$1.6x10^5$	$1.2x10^3$	0.5	0.05	α =135 $\beta\gamma$=240	Location 27, spot on concrete floor

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm²)		End Window (mR/h)		Smear Results (dis/min-100 cm²)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
Grids 12 through 19 (cont'd.)				4.9×10^4	BKGD[c]	0.1	BKGD	BKGD	Location 29, spot on concrete floor
				1.5×10^4	BKGD	0.07	BKGD	BKGD	Location 32, spot on concrete floor
				1.0×10^3	BKGD	BKGD	BKGD	BKGD	Location 33, spot on concrete floor
				6.8×10^5	5.8×10^3	7	BKGD	α =510[e] $\beta\gamma$=850	Location 34, spot of yellow residue on a steel pump valve flange
				2.1×10^3	BKGD	BKGD	BKGD	BKGD	Location 35, spot on concrete floor
				BKGD	NA	NA	BKGD	BKGD	Rest of survey was BKGD
Grids 20 and 21				6.8×10^5	BKGD	2.0	0.05	NST[f]	Location 38, spot on concrete floor
				2.4×10^5	BKGD	0.08	BKGD	BKGD	Location 40, spot on concrete floor
				1.0×10^6	BKGD	5	0.2	BKGD	Location 41, spot on concrete floor

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm^2)		End Window (mR/h)		Smear Results (dis/min-100 cm^2)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
Grids 20 and 21 (cont'd.)				2.6×10^4	BKGD[c]	0.09	BKGD	BKGD	Location 43, spot on concrete floor
				2.6×10^4	BKGD	BKGD	BKGD	BKGD	Rest of survey, general contamination on about 30% of the floor
Grids 22 through 25				1.7×10^3	BKGD	BKGD	BKGD	BKGD	Location 47, spot on concrete floor
				BKGD	NA[d]	NA	BKGD	BKGD	Rest of survey was BKGD
Overheads				1.4×10^3	BKGD	BKGD	BKGD	BKGD	Location 81, spot on steel overhead beam
				2.1×10^3	BKGD	BKGD	BKGD	$\alpha = 10$[e] $\beta\gamma = 33$	Location 83, spot on steel overhead beam
				6.9×10^2	BKGD	BKGD	BKGD	$\alpha = 6$ $\beta\gamma$=BKGD	Location 84, spot on steel overhead beam
				BKGD	NA	NA	BKGD	BKGD	Rest of survey was BKGD
Acct. Storage	90	95	NS[b]	BKGD	NA	NA	BKGD	BKGD	
Dock Area	90	95	NS	BKGD	NA	NA	BKGD	BKGD	
Entrance Corridor	90	95	NS	BKGD	NA	NA	BKGD	BKGD	

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm²)		End Window (mR/h)		Smear Results (dis/min-100 cm²)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
2nd Level of Main Room	80	75	NS[b]	6.6×10^4	BKGD[c]	7	BKGD	α =56[e] $\beta\gamma$=220	Location 102, area on top of stainless steel Nitric Acid Tank-1
				3.4×10^5	BKGD	1.5	BKGD	α =28 $\beta\gamma$=96	Location 107, spot on steel top of Nitric Acid Tank-1
				1.6×10^4	BKGD	0.3	BKGD	α =25 $\beta\gamma$=86	Location 103, spot on steel floor beam
				3.4×10^5	BKGD	3	BKGD	α =640 $\beta\gamma$=780	Location 104, spot on steel floor beam
				6.8×10^4	BKGD	0.4	BKGD	α =130 $\beta\gamma$=260	Location 105, spot on steel floor beam
				BKGD	NA[d]	NA	BKGD	BKGD	Rest of survey was BKGD
2nd Level Lab	90	80	0.0047						Air Sample 5
				7.9×10^4	4.0×10^3	0.5	BKGD	α =12 $\beta\gamma$=BKGD	Location 119, area in soapstone sink in lab
				BKGD	NA	NA	BKGD	BKGD	Rest of survey was BKGD
2nd Level Change Room	90	80	NS	BKGD	NA	NA	BKGD	BKGD	

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm^2)		End Window (mR/h)		Smear Results (dis/min-100 cm^2)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
3rd Level	90	NA	0.0025						Air Sample 2, at sound-proof booth
				6.5×10^4	4.0×10^3	1.2	BKGD[c]	α =BKGD[e] $\beta\gamma$=44	Location 132, spot on stainless steel Kelly-1
				1.4×10^6	4.6×10^3	5	0.15	BKGD	Location 133, spot in stainless steel Kelly-1
				3.3×10^4	BKGD	0.05	BKGD	BKGD	Location 134, spot on steel floor beam
				6.8×10^5	2.3×10^3	0.15	BKGD	α =42 $\beta\gamma$=19	Location 135, spot on stainless steel Kelly-1
				BKGD	NA[d]	NA	BKGD	BKGD	Rest of survey was BKGD
4th Level Catwalk	100	No walls	NS[b]	BKGD	NA	NA	BKGD	BKGD	Walkway over Kellys
Roof	100	No Walls	0.0014						Air Sample 3
				3.4×10^4	BKGD	0.06	BKGD	BKGD	Location 147, spot on galvanized steel roof vent, equated to ^{226}Ra

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm^2)		End Window (mR/h)		Smear Results (dis/min-100 cm^2)	Comments
	Floor	Wall		Beta	Alpha	Contact	1 meter		
Roof (cont'd.)				6.0×10^3	BKGD[c]	BKGD	BKGD	BKGD	Location 148, spot on gravel, dirt, and tar roof floor, equated to ^{226}Ra.
				1.8×10^3	BKGD	NA[d]	BKGD	BKGD	Rest of survey, general contamination equated to ^{226}Ra.
Interior of Tanks:									
Filtered Liquor Tank-5	90	95	NS[b]	3.4×10^5	2.3×10^3	1.0	BKGD	BKGD	Location 164, spot inside steel crossover pipe
				2.1×10^3	BKGD	BKGD	BKGD	α =9[e] $\beta\gamma$=BKGD	Location 157, spot on steel wall
				1.3×10^5	2.3×10^3	BKGD	BKGD	NST	Ring of contamination along walls, 2 ft below top of tank
				2.1×10^3	BKGD	BKGD	BKGD	BKGD	Rest of survey, general contamination on floor and walls
Filtered Liquor Tank-6	90	95	NS	3.4×10^5	BKGD	1.8	0.07	BKGD	Location 174, spot inside steel inlet pipe

TABLE 1

DATA SHEET OF ROOM SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window (mR/h) Contact	1 meter	Smear Results (dis/min-100 cm²)	Comments
Filtered Liquor Tank-6 (cont'd.)				7.5×10^3	1.7×10^3	0.06	BKGD[c]	BKGD	General contamination on steel walls
				6.9×10^2	BKGD	0.06	BKGD	BKGD	General contamination on steel floor
				3.4×10^2	BKGD	BKGD	BKGD	NST[f]	General contamination on center stirrer.
Nitric Acid Tank-1	90	95	NS[b]	BKGD	NA[d]	NA	BKGD	BKGD	
Kelly Feed Tank-1	90	95	NS	5.5×10^3	BKGD	0.06	BKGD	BKGD	Location 191, spot steel pipe
				1.7×10^3	BKGD	BKGD	BKGD	α =BKGD[e] $\beta\gamma$=27	Location 188, spot steel wall
				1.7×10^3	BKGD	BKGD	BKGD	BKGD	General contamination on rest of walls
				6.9×10^2	BKGD	BKGD	BKGD	NST	General contamination on center stirrer
				BKGD	NA	NA	BKGD	BKGD	Rest of survey (floors) was BKGD

FOOTNOTES FOR TABLE 1

[a]The Beta Mode Direct Readings and Alpha Mode Direct Readings are taken with PAC-4G-3 instruments (see Appendix 1). The beta mode detects both electromagnetic and particulate radiation. If an area indicated an instrument reading higher than background, a beta mode reading was obtained. The instrument was then switched to the alpha mode, and a reading of the alpha contamination was obtained. In the alpha mode, the instrument only responds to particles with high specific ionization, such as alpha particles. The beta mode readings are compensated for any alpha contribution by subtracting the alpha mode reading from the beta mode reading.

[b]NS (Not Selected) Locations of air samples were chosen on a selected basis throughout the areas surveyed. "NS" indicates that the room or area was not selected for an air sample.

[c]BKGD (Background) The following are the instrument background readings:

	Beta Mode	Alpha Mode
Floor Monitor	1500-2000 cts/min-325 cm^2	0-50 cts/min-325 cm^2
PAC-4G-3	150-200 cts/min-51 cm^2	0-50 cts/min-51 cm^2
PC-5 Counter	40.0±1.4 cts/min*	0.2±0.1cts/min*
10-Wire	443.0±4.7 cts/min*	5.2±0.5 cts/min*

GM End Window Detector read 0.02 to 0.03 mR/h at 1 m above floor.

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

[e]α =Alpha
$\beta\gamma$=Beta-gamma (The beta-gamma readings are compensated for any alpha contribution by subtracting the alpha reading from the beta-gamma reading.)

[f]NST (No Smear Taken)

[g]NRR (No Reading Recorded).

*One standard deviation due to counting statistics.

TABLE 2

LOCATIONS WHERE RESIDUAL CONTAMINATION EXCEEDED ACCEPTABLE LIMITS[a,b]

Area	Location Number	Estimated Area of Contamination (cm^2)	Maximum PAC Reading (dis/min-100 cm^2)		Contact GM Reading (mR/h)	Smear Results (dis/min-100 cm)	
			Beta-Gamma	Alpha		Beta-Gamma	Alpha
1st Level of Main Room Grids 1 through 9	1	500	$1.2x10^4$	BKGD[c]	0.1	NST	NST
	2	500	$8.2x10^3$	BKGD	0.05	NST	NST
	12	$1.2x10^5$	$2.7x10^4$	BKGD	BKGD	BKGD	BKGD
	13	500	$1.1x10^4$	BKGD	BKGD	BKGD	BKGD
	14	500	$1.6x10^4$	BKGD	BKGD	BKGD	BKGD
	15	500	$1.1x10^4$	BKGD	BKGD	BKGD	BKGD
	16	500	$1.1x10^4$	BKGD	BKGD	BKGD	BKGD
	20	500	$1.6x10^4$	BKGD	0.05	BKGD	BKGD
Grids 10 and 11	22	500	$2.3x10^4$	$1.2x10^3$	0.1	BKGD	BKGD
	23	500	$3.4x10^4$	$1.2x10^3$	0.05	BKGD	BKGD
	24	500	$3.4x10^4$	$1.2x10^3$	0.2	BKGD	BKGD
	26	500	$1.6x10^5$	$2.9x10^3$	0.4	18	6

TABLE 2 (continued)

Area	Location Number	Estimated Area of Contamination (cm^2)	Maximum PAC Reading (dis/min-100 cm^2)		Contact GM Reading (mR/h)	Smear Results (dis/min-100 cm)	
			Beta-Gamma	Alpha		Beta-Gamma	Alpha
Grids 12 through 19	27	500	$1.6x10^5$	$1.2x10^3$	0.5	240	135
	29	500	$4.9x10^4$	BKGD	0.1	BKGD	BKGD
	32	500	$1.5x10^4$	BKGD	0.07	BKGD	BKGD
	34	500	$6.8x10^5$	$5.8x10^3$	7	850	510
Grids 20 and 21	38	500	$6.8x10^5$	BKGD	2.0	NST	NST
	40	500	$2.4x10^5$	BKGD	0.08	BKGD	BKGD
	41	500	$1.0x10^6$	BKGD	5	BKGD	BKGD
	43	500	$2.6x10^4$	BKGD	0.09	BKGD	BKGD
	General	$3.5x10^5$	$2.6x10^4$	BKGD	BKGD	BKGD	BKGD
2nd Level of Main Room	102	$7.2x10^4$	$6.6x10^4$	BKGD	7	220	56
	103	500	$1.6x10^4$	BKGD	0.3	86	25
	104	500	$3.4x10^5$	BKGD	3	780	640
	105	500	$6.8x10^4$	BKGD	0.4	260	130
	107	500	$3.4x10^5$	BKGD	1.5	96	28
2nd Level Lab	119	$2.1x10^3$	$7.9x10^4$	$4.0x10^3$	0.5	BKGD	12

Table 2 (continued)

Area	Location Number	Estimated Area of Contamination (cm^2)	Maximum PAC Reading (dis/min-100 cm^2)		Contact GM Reading (mR/h)	Smear Results (dis/min-100 cm)	
			Beta-Gamma	Alpha		Beta-Gamma	Alpha
3rd Level	132	500	$6.5x10^4$	$4.0x10^3$	1.2	44	BKGD
	133	500	$1.4x10^6$	$4.6x10^3$	5	BKGD	BKGD
	134	500	$3.3x10^4$	BKGD	0.05	BKGD	BKGD
	135	500	$6.8x10^5$	$2.3x10^3$	0.15	19	42
Roof	147	500	$3.4x10^4$[d]	BKGD	0.06	BKGD	BKGD
	148	500	$6.0x10^3$[d]	BKGD	BKGD	BKGD	BKGD
	General	$7.5x10^6$	$1.8x10^3$[d]	BKGD	BKGD	BKGD	BKGD
Interior of Tanks:							
Filtered Liquor Tank-5	164	500	$3.4x10^5$	$2.3x10^3$	1.0	BKGD	BKGD
	General	$3.6x10^4$	$1.3x10^5$	$2.3x10^3$	BKGD	BKGD	BKGD
Filtered Liquor Tank-6	174	500	$3.4x10^5$	BKGD	1.8	BKGD	BKGD
	General	$4.4x10^5$	$7.5x10^3$	$1.7x10^3$	0.06	BKGD	BKGD
Kelly Feed Tank-1	191	500	$5.5x10^3$	BKGD	0.06	BKGD	BKGD

Table 2
(cont'd.)

FOOTNOTES

[a]Locations are indicated in Table 1 and Figures 1 through 9.

[b]The surface contamination limits for uranium (radium for the roof) as given in the Draft ANSI Standard N13.12 and the average and maximum radiation levels at 1 cm as given in the NRC Guidelines were used as the standards for "acceptable levels of contamination."

[c]BKGD = Background

[d]dis/min-100 cm^2, equated to ^{226}Ra plus daughters

TABLE 3

RADON DETERMINATIONS

Sample Number	Location	Figure	dis/min-m^3	pCi/ℓ	WL[a]
1	1st Level	1b	574	0.26	0.0026
2	3rd Level	3	560	0.25	0.0025
3	Roof	5	313	0.14	0.0014
4	1st Level	1b	1375	0.61	0.0061
5	2nd Level	2	1055	0.47	0.0047

Example Calculation: Air Sample 1, First Level

$$\frac{574 \text{ dis/min}}{m^3} \times \frac{1 \text{ pCi}}{2.22 \text{ dis/min}} \times \frac{m^3}{10^3 \ \ell} \times \frac{\text{WL}}{100 \text{ pCi}/\ell} = 0.0026 \text{ WL}$$

[a]A Working Level (WL) is defined as any combination of short-lived radon-decay products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. The numerical value of the WL is derived from the alpha energy released by the total decay through RaC' of the short-lived radon-daughter products, RaA, RaB, and RaC at radioactive equilibrium with 100 pCi of ^{222}Rn per liter of air.

TABLE 4

SOIL AND MATERIAL SAMPLE WEIGHTS
(grams)

Sample Number[a]	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
4-S1	111.8	95.5	54.7	40.7
4-S2	99.9	99.0	23.5	75.2
4-S3	-	-	100	-
4-S4	-	-	100	-
4-S5	-	-	100	-
4-S6-A	380.0	362.6	314.5	46.0
4-S6-B	782.2	750.2	527.0	213.4
4-S6-C	1713.7	1639.9	737.8	891.0
4-S6-D	2149.0	2065.8	776.6	1280.0
4-S7-A	705.0	636.5	260.0	367.0
4-S7-B	793.4	732.7	389.7	341.7
4-S7-C	963.8	902.6	498.0	400.3
4-S7-D	3420.0	3236.8	984.7	2242.1
4-S8-A	820.7	649.3	352.0	289.0
4-S8-B	435.6	402.0	219.5	180.3
4-S8-C	782.0	704.1	509.3	193.2
4-S8-D	2930.3	2667.5	783.9	1865.2
4-S9-A	970.8	875.9	584.3	290.0
4-S9-B	505.6	454.4	298.1	154.2
4-S9-C	1180.0	1076.7	600.6	473.5
4-S9-D	3951.2	3376.0	1460.4	1913.0
4-S10-A	493.1	330.0	258.5	67.4
4-S10-B	499.6	354.6	273.3	76.7
4-S10-C	522.5	399.3	54.6	143.5
4-S10-D	2044.5	1761.8	710.1	1043.4
4-S11-A	528.6	489.0	144.3	342.9
4-S11-B	437.8	407.5	99.2	305.8
4-S11-C	526.8	470.0	132.8	335.6
4-S11-D	2200.9	2032.1	427.0	1600.5

TABLE 4
(cont'd.)

Sample Number	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
4-S12-A	587.4	474.3	450.7	22.3
4-S12-B	450.8	373.2	365.1	6.1
4-S12-C	876.2	740.9	609.6	122.5
4-S12-D	1524.8	1293.4	837.2	442.7
4-S13-A	306.4	272.1	228.7	41.7
4-S13-B	428.4	390.9	245.5	143.3
4-S13-C	576.1	513.0	370.4	140.5
4-S13-D	1635.5	1448.1	902.9	529.1
4-S14-A	592.0	500.8	460.0	35.5
4-S14-B	878.0	756.3	404.2	342.4
4-S14-C	763.3	659.0	526.8	121.0
4-S14-D	2473.6	2108.6	898.0	1156.3
4-S15-A	833.9	668.1	364.0	303.0
4-S15-B	691.1	554.3	380.0	156.0
4-S15-C	808.3	650.4	488.0	142.0
4-S15-D	1751.7	1385.8	883.0	496.0

*Samples 4-S1 and 4-S2 consisted of gravel from the roof of Building 55; samples 4-S3 through 4-S5 consisted of material/chemical samples related to current operations; and the remainder of the samples were soil samples collected from the grounds around the building.

TABLE 5

GAMMA-RAY SPECTRAL AND URANIUM-FLUOROMETRIC ANALYSES OF SOIL AND MATERIAL SAMPLES

	Ge(Li) Spectra pCi/g received wt±σ[a]				
Sample Number	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium μg/g±σ[b]	Uranium pCi/g±σ[c]
4-S1 (Roof gravel)	1.8 ±0.3	1.8 ±0.9	14.6 ±0.7* [16 ±0.8]	200 ±10	136 ± 7
4-S2 (Roof gravel)	1.0 ±0.2	2.3 ±0.7	15.1 ±0.8* [14.4 ±0.7]	68 ±3	46 ± 2
4-S3 (Disodium phosphate before filtration)	ND[d]	1.4 ±0.6	0.7 ±0.3* [0 ±0.1]	128 ±6	87 ± 4
4-S4 (Disodium phosphate after filtration)	ND[d]	2.4 ±1.8	0.64±0.15* [0.63±0.4]	270 ±10	184 ± 7
4-S5 (Ground Phosphate Rock)	0 ±0.6	1.3 ±0.3	36.0 ±1.8* [39.0 ±2.0]	117 ±12 (51 ±2)	79.6 ± 8.0 (35 ± 1)
4-S6-A	≤ 0.02	0.35±0.07* (0.49±0.10)	0.18±0.03* (0.13)	0 ±0.3	0 ± 0.2
4-S6-B				5.0±0.3	3.4 ± 0.2
4-S6-C				0 ±0.3	0 ± 0.2
4-S6-D				0.5±0.3	0.3 ± 0.2
4-S7-A	0.22±0.03	0.27±0.06	1.53±0.08	1.1±0.3	0.75± 0.2
4-S7-B				2.1±0.4	1.4 ± 0.3
4-S7-C				1.1±0.4	0.75± 0.3
4-S7-D				2.0±0.6	1.4 ± 0.4
4-S8-A	0.17±0.02* (0.21±0.08)	0.71±0.10	7.96±0.40* (8.48±0.42)	129 ±7 (227 ±23)	87.7 ± 4.8 (154 ±15)
4-S8-B				14.8±0.4	10.2 ± 0.3
4-S8-C				14.5±0.7	9.9 ± 0.5
4-S8-D				28 ±1	19 ± 0.7
4-S9-A	0.15±0.06	0.97±0.19	18.1 ±0.91	78 ±4 (62 ±6)	53 ± 3 (42 ± 4)
4-S9-B				43 ±2	29 ± 1
4-S9-C				30 ±2	20 ± 1
4-S9-D				24 ±2	16 ± 1

Table 5
(cont'd.)

Sample Number	Ge(Li) Spectra (pCi/g received wt±σ)[a] ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium µg/g±σ[b]		Uranium pCi/g±σ[c]	
4-S10-A	2.02±0.08	1.14±0.17	1.30±0.12	3.4	±0.4	2.3	± 0.3
4-S10-B				1.9	±0.4	1.3	± 0.3
4-S10-C				1.6	±0.4	1.1	± 0.3
4-S10-D				1.2	±0.4	0.82	± 0.3
4-S11-A	1.68±0.08	0.77±0.11	2.63±0.13	6.4	±0.4	4.4	± 0.3
4-S11-B				6.5	±0.4	4.4	± 0.3
4-S11-C				6.2	±0.4	4.2	± 0.3
4-S11-D				3.5	±0.4	2.4	± 0.3
4-S12-A	1.20±0.07	1.08±0.18	0.74±0.10	1.3	±0.3	0.88	± 0.2
4-S12-B				1.0	±0.2	0.7	± 0.1
4-S12-C				1.3	±0.2	0.88	± 0.1
4-S12-D				1.2	±0.2	0.82	± 0.1
4-S13-A	0.52±0.05	0.66±0.13	1.41±0.08	3.6	±0.4	2.4	± 0.3
4-S13-B				2.3	±0.3	1.6	± 0.2
4-S13-C				2.9	±0.4	2.0	± 0.3
4-S13-D				2.3	±0.3	1.6	± 0.2
4-S14-A	1.35±0.08	1.51±0.20	1.39±0.13	3.3	±0.3	2.2	± 0.2
4-S14-B				1.5	±0.3	1.0	± 0.2
4-S14-C				2.8	±0.4	1.9	± 0.3
4-S14-D				2.8	±0.4	1.9	± 0.3
4-S15-A	1.37±0.07	0.89±0.12	1.76±0.09	4.4	±1.8	3.0	± 1.2
4-S15-B				3.4	±0.2	2.3	± 0.1
4-S15-C				2.2	±0.4	1.5	± 0.3
4-S15-D				2.5	±0.3	1.7	± 0.2

*NOTE: Best values are listed. Results of other counting measurements are listed in parentheses. Results of radium by radon emanation are listed in brackets.

(a) One standard deviation due to counting statistics.
(b) Data results from LFE Environmental Analysis Laboratories.
(c) ANL conversion per Appendix 5.
(d) Not detected.

TABLE 6

BACKGROUND SOIL SAMPLE DATA[a]
Cesium-137, Thorium-232, Natural Uranium in Soil, 1978
(Concentrations in pCi/g)

Date Collected	Location	Cesium-137	Thorium-232	Uranium[b] (natural)
June 23	Argonne Area[c]	0.8±0.2	0.26±0.02	1.0±0.1
June 23	Argonne Area	0.3±0.1	0.60±0.04	2.2±0.2
June 23	Argonne Area	1.3±0.3	0.40±0.03	1.3±0.1
June 23	Argonne Area	1.2±0.3	0.38±0.03	1.5±0.1
June 23	Argonne Area	1.2±0.3	0.38±0.03	1.7±0.1
October 17	Argonne Area	3.0±0.7	0.18±0.02	1.2±0.1
October 17	Argonne Area	1.3±0.4	0.36±0.04	1.0±0.1
October 17	Argonne Area	1.1±0.3	0.40±0.04	1.2±0.3
October 17	Argonne Area	1.5±0.4	0.48±0.04	1.3±0.2
October 17	Argonne Area	1.0±0.3	0.40±0.02	1.5±0.2
	Average	1.3±0.4	0.38±0.07	1.4±0.2
June 16	Naperville, IL	1.2±0.3	0.53±0.03	1.6±0.2
June 20	Channahon, IL	1.1±0.3	0.36±0.02	1.5±0.1
June 20	Morris, IL	1.2±0.3	0.27±0.03	1.2±0.2
June 20	Starved Rock State Pk., IL	0.9±0.3	0.19±0.02	0.6±0.1
June 21	Willow Springs, IL	0.9±0.3	0.31±0.03	1.4±0.1
October 19	McKinely Wds. State Pk., IL	1.3±0.4	0.39±0.05	1.4±0.3
October 19	Dresden Lock and Dam, IL	1.6±0.5	-	1.3±0.1
October 20	Romeoville, IL	2.9±0.7	0.42±0.04	2.2±0.3
October 20	Lemont, IL	0.8±0.3	0.37±0.04	1.1±0.1
October 20	McGinnis Slough, IL	1.3±0.4	0.37±0.04	1.6±0.1
October 20	Sagnashkee Slough, IL	1.1±0.3	0.37±0.04	1.8±0.1
	Average	1.3±0.3	0.36±0.06	1.4±0.2

[a]These results are from "Environmental Monitoring at Argonne National Laboratory: Annual Report for 1978" (ANL-79-24) by N. W. Golchert, T. L. Duffy, and J. Sedlet

[b]Uranium (natural) includes 2.26% ^{235}U, and 48.87% each of ^{238}U and ^{234}U. This implies that at equilibrium the concentrations of ^{226}Ra in each sample would be 48.87% of the concentration shown for uranium (natural).

[c]All samples marked "Argonne Area" were collected at Argonne National Laboratory near Lemont, IL, southwest of Chicago.

TABLE 7

ESTIMATED VOLUME, MASS, AND ACTIVITY OF MATERIAL THAT COULD BE GENERATED BY REMEDIAL ACTION[a]

Type of Material	Estimated Volume (m^3)	Estimated Mass (kg)	Estimated Activity (μCi)
Concrete floors			
Option A (ρ=0.1 g/cm^3)[b]	5.7×10^{-1}	5.7×10^{1}	6.0×10^{1}
Option B (ρ=2.35 g/cm^3)	2.4	5.7×10^{3}	6.0×10^{1}
Steel			
Option A (ρ=0.1 g/cm^3)	1.4×10^{-1}	1.4×10^{1}	6.7×10^{1}
Option B (ρ=7.8 g/cm^3)	1.8×10^{-1}	1.4×10^{3}	6.7×10^{1}
Soapstone (ρ=2.8 g/cm^3)	1.1×10^{-2}	3.0×10^{1}	7.4×10^{-1}
Roof gravel, dirt, tar (ρ=2.0 g/cm^3)	1.9×10^{1}	3.8×10^{4}	6.1×10^{1}
Soil around sample locations 4-S8 & 9 (ρ=2.0 g/cm^3)	4.0	8.0×10^{3}	7.3×10^{1}
TOTAL			
Option A	2.4×10^{1}	4.6×10^{4}	2.6×10^{2}
Option B	2.6×10^{1}	5.4×10^{4}	2.6×10^{2}

[a] See text for assumptions upon which estimates are based.

[b] The assumed density for the purpose of calculating mass of material.

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 alpha counter which has a gas-proportional probe, 51 cm^2 (PAC-4G-3) or 325 cm^2 (FM-4G) in area, with a thin double-aluminized Mylar window (~0.85 mg/cm^2).

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

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; however, the response (in mR/h) to beta radiation is nonconservative by a factor of up to about four relative to a determination of mrad/h through 7 mg/cm^2. 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.

II. SMEAR-COUNTING INSTRUMENTATION

The 10-wire instrument consists of a gas-flow proportional probe (ANL design) which uses an Eberline Mini Scaler Model MS-2. The double-aluminized Mylar probe (400 cm^2) uses P-10 (90% argon and 10% methane) as the counting gas. This system consists of two Mini Scalers and two probes. One is used for counting in the alpha mode; the other is used in the beta mode. The metal smear holder has been machined to hold ten smears. The probe is placed over the smears and a count is taken.

APPENDIX 1
(cont'd.)

All smears of contaminated areas are counted in a Nuclear Measurements Corporation PC-5 Gas-Flow Proportional Counter (PC counter) using a double-aluminized Mylar spun top. The Mylar spun top is placed over nonconducting media such as paper to negate the dielectric effect. This counter also uses P-10 counting gas. Smears are counted in both the alpha and beta modes of the detector. These instruments are calibrated using ^{239}Pu and ^{90}Sr-^{90}Y NBS-traceable sources.

III. AIR-SAMPLING DEVICE

The air samples were collected with a commercial vacuum cleaner modified at ANL. The air was drawn at a flow rate of 40 m^3/h. The collection medium consisted of a 200 cm^2 sheet of Hollingsworth-Vose (HV-70-0.23 mm) filter paper. The collection efficiency at this flow rate for 0.3-micron particles is about 99.9%.

IV. GAMMA-SPECTRAL INSTRUMENTATION

A Nuclear Data Multichannel Analyzer Model ND-100 with a 7.6-cm-diameter by 7.6-cm-long NaI(Tℓ) crystal was used to determine the gamma spectrum. This instrument was calibrated with NBS-traceable sources. Samples of contaminated areas were counted with the analyzer, and the radionuclides of contamination were determined.

V. INSTRUMENTATION USED IN SURVEY

	Inventory Number	Probe Area (cm^2)	Window Thickness (mg/cm^2)
Eberline Floor Monitor FM-4G using a PAC-4G-3	181501	325	~0.85
Eberline Floor Monitor FM-4G using a PAC-4G-3	183413	325	~0.85
PAC-4G-3	183414	51	~0.85
PAC-4G-3	183415	51	~0.85
PAC-4G-3	183416	51	~0.85
PAC-4G-3	184339	51	~0.85
PAC-4G-3	184340	51	~0.85
PAC-4G-3	184341	51	~0.85
Eberline 530 with HP-190 Beta-Gamma End Window Probe	184576	-	~7
Nuclear Measurements Corp. PC-5 2π Internal Gas-Flow Counter	184065	-	~0.85

APPENDIX 1
(cont'd.)

	Inventory Number	Probe Area (cm^2)	Window Thickness (mg/cm^2)
Argonne National Laboratory 10-Wire Flat-Plate Gas-Flow Proportional Detector with Eberline Mini Scaler MS-2	184342 & 184343	400	~0.85
Argonne National Laboratory Filter Queen Air Sampler using HV-70 filter media	-	-	-
Nuclear Data Multichannel Analyzer Model ND-100 with 7.6 cm dia x 7.6 cm NaI(Tℓ) crystal	184764	-	-

VI. AVERAGE INSTRUMENT BACKGROUND READINGS

Instrument	Alpha Mode (cts/min)	Beta Mode (cts/min)	1 m above floor
Eberline Floor Monitor FM-4G using PAC-4G-3			
181501	0-50	1500-2000	
183413	0-50	1500-2000	
Eberline PAC-4G-3			
183414	0-50	150-200	
183415	0-50	150-200	
183416	0-50	150-200	
184339	0-50	150-200	
184340	0-50	150-200	
184341	0-50	150-200	
Eberline 530 with HP-190 Beta-Gamma End Window Probe			0.02-0.03 mR/h
Nuclear Measurements Corporation PC-5 2π Internal Gas-Flow Counter	0.2±0.1[b]	40.0±1.4[b]	
Argonne National Laboratory 10-Wire Flat-Plate Gas-Flow Proportional Detector with Eberline Mini Scaler MS-2	5.2±0.5	443.0±4.7	

[a]Background readings were initially taken in the mobile laboratory and rechecked throughout the various areas while surveying.

[b]One standard deviation due to counting statistics.

APPENDIX 2

CONVERSION FACTORS

I. INSTRUMENTATION

The factors used to convert the instrument readings into units of disintegrations per minute per 100 cm^2 (dis/min-100 cm^2) alpha 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
dis/min per cts/min for ^{239}Pu	2	-	2	-
dis/min per cts/min for ^{90}Sr-^{90}Y	-	2	-	2
dis/min per cts/min for normal uranium	5.9	3.5	-	-
dis/min α per cts/min for ^{226}Ra plus daughters	1.6	4.7	-	-

B. Derivation of Conversion Factors

- Floor Monitor (FM-4G)

  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.)

A 3.2 x 3.2 x 0.3-cm normal-uranium plate (used as a source of uranium alpha emissions) was counted in the well of a 2π Internal Gas-Flow Counter (PC Counter) with the source leveled to an apparent 2π geometry. As previously stated, this instrument was calibrated using ^{239}Pu NBS-traceable sources. The alpha reading was 4.7×10^4 cts/min, or $4.7 \times 10^4 \div 0.50 = 9.4 \times 10^4$ dis/min alpha with the PC counter.

The same uranium source, when counted in the alpha mode with the PAC instrument, was found to be 1.6×10^4 cts/min at contact. The conversion factor for cts/min to dis/min for the PAC instrument is $9.4 \times 10^4 \div 1.6 \times 10^4 = 5.9$ dis/min alpha per cts/min alpha.

The same normal uranium source covered with two layers of conducting paper, each 6.65 mg/cm^2 to absorb the alpha emissions, was counted for composite beta and gamma emissions in the PC counter; however, no provision was made for back-scatter. The composite beta-gamma count was 5.2×10^5 cts/min, or $5.2 \times 10^5 \div 0.50 = 1.04 \times 10^6$ dis/min beta-gamma.

When the covered normal uranium source was counted in the beta mode of the PAC-4G-3, the count rate was 3.0×10^5 cts/min. This indicates a conversion factor of $1.04 \times 10^6 \div 3.0 \times 10^5 = 3.5$ dis/min beta-gamma per cts/min beta-gamma.

A similar method was used to determine the conversion factors for ^{226}Ra plus daughters.

II. SMEAR COUNT

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

A. Conversion Equation (Alpha)

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

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

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

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

If the energies of the isotope were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the (waf) of ^{239}Pu (0.713) was used.

The (waf) for normal-uranium alphas is 0.54.

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

APPENDIX 2
(Cont'd.)

E. 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 is 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 normal-uranium betas is 0.85.

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

APPENDIX 3

RADON-DETERMINATION CALCULATIONS

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

I. RADON CONCENTRATIONS BASED ON RaC' RESULTS

The following postulates are assumed in deriving the radon-222 (^{222}Rn) concentrations as 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 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, which is determined from RaC'.

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. No long-lived alpha emitters are present, as evidenced by the final count.

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

APPENDIX 3
(Cont'd.)

II. EQUATIONS USED TO DERIVE AIR CONCENTRATIONS

The activity present at the end of the sampling period is determined by the equation:

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

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

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

t = Time interval (min) from end of sampling period to counting interval (usually $\sim$ 100 min)

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

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

Concentration is determined by the equation:

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

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 min)

t_s = Length of sampling time (min)

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

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

APPENDIX 3
(Cont'd.)

III. EXAMPLE CALCULATION:

Data obtained from air sample 1, collected in the first level main room, has been used to illustrate the application of the equations for determining activity and concentration.

$$A_o = \frac{778}{\exp \frac{-0.693 \times 100}{36}} = 5335 \text{ dis/min}$$

$$C = \frac{\frac{0.693}{5335 \times 36}}{40/60} \times \frac{1}{1 - \exp \frac{-0.693 \times 40}{36}} = 287 \text{ dis/min-m}^3$$

Since we assume that half of the radon progeny is not adhered to the airborne particulates, the above concentration is then multiplied by a factor of two to determine the actual concentration:

C actual = C measured x progeny correction factor

$$= 287 \text{ dis/min-m}^3 \times 2 = 574 \text{ dis/min-m}^3$$

The resultant concentration is 574 dis/min-m^3.

APPENDIX 4

SOIL ANALYSIS PROCEDURE FOR TOTAL URANIUM AND GAMMA-EMITTING NUCLIDES*

A 60-milliliter volume of the received soil was counted in a petri dish for 500 minutes on a Ge(Li) detector over the energy range 0-1.5 MeV. This corresponded to 60-100 g of soil, depending upon bulk soil density. Positive photopeaks above instrument background were converted to dis/min using a line efficiency curve based upon a National Bureau of Standards Multi-Gamma standard. The natural thorium-232 and radium-226 decay chains were calculated using the 0.910-MeV actinium-228 and 0.609-MeV bismuth-214 photopeaks, respectively. Cesium-137 is reported for each sample as a representative gamma emitter. Potassium-40 was observed on all soil samples, as expected, but was not calculated or reported.

One gram of the soil sample was ashed and dissolved in HF-HNO_3 for the total uranium analysis. A 100-λ aliquot of the dissolved sample was fused with 98% NaF-2% LiF and the fluorescence determined using a Jarrell-Ash fluorometer. A quenching factor was determined for each sample by using an internal spike.

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*The procedures outlined in this appendix are those used by LFE Environmental Analysis Laboratories to analyze the soil samples collected near Building 55 of the Chemicals Group, Olin Corporation.

APPENDIX 5

CALCULATION OF NORMAL-URANIUM SPECIFIC ACTIVITY

Radioactive half-lives of ^{234}U, ^{235}U, and ^{238}U, as well as the percent abundance for each isotope, were obtained as current best values from the "Table of Isotopes"--7th Edition by C.M. Lederer and V. S. Shirley, 1978. The values used are:

Isotope	Half-life (years)	Abundance (atom %)	Atomic Weight (grams)	Abundance (wt %)
^{234}U	2.446×10^5	0.0054	234.04	0.0053
^{235}U	7.038×10^8	0.720	235.04	0.711
^{238}U	4.4683×10^9	99.275	238.05	99.284
		100.0004		100.0003

Note that the abundance totals 100.0003%. Since it cannot be determined which isotope(s) are in error, the calculations are made with the 0.0003% error unaccounted for.

Specific activity, or activity per unit mass, is determined by the equation:

$$SpA = \lambda N$$

where: SpA = Specific Activity

$\lambda = \ell n2/t_{\frac{1}{2}}$

N = Number of radioactive atoms per unit mass

$$= \frac{\text{Avogadro's Number}}{\text{gram atomic weight}}$$

Avogadro's Number = 6.022×10^{23}

$t_{\frac{1}{2}}$ = Half-life in years (a)

Therefore:

$$SpA = (\ell n2)N/t_{\frac{1}{2}}$$

$$= \frac{\ell n2 \times 6.022 \times 10^{23}}{t_{\frac{1}{2}}(a) \times 5.2596 \times 10^5 \frac{\text{min}}{a} \times \text{gram atomic weight}} \quad \text{in dis/min-gram.}$$

APPENDIX 5
(cont'd.)

For ^{234}U, the specific activity would be:

$$SpA\ ^{234}U = \frac{\ell n2 \times 6.022 \times 10^{23}}{2.446 \times 10^{5} \times 5.2596 \times 10^{5} \times 2.34 \times 10^{2}}$$

$$= 1.386 \times 10^{10} \text{ dis/min-gram}$$

$$= 6.245 \times 10^{3} \text{ pCi/}\mu g \times 0.0053\% = 0.331 \text{ pCi/}\mu g \text{ of normal U}$$

where 2.2×10^{6} dis/min-gram = 1 pCi/μg

For ^{235}U, the specific activity would be:

$$SpA\ ^{235}U = \frac{\ell n2 \times 6.022 \times 10^{23}}{7.038 \times 10^{8} \times 5.2596 \times 10^{5} \times 2.3504 \times 10^{2}}$$

$$= 4.798 \times 10^{6} \text{ dis/min-gram}$$

$$= 2.161 \text{ pCi/}\mu g \times 0.711\% = 1.54 \times 10^{-2} \text{ pCi/}\mu g \text{ of normal U}$$

For ^{238}U, the specific activity would be:

$$SpA\ ^{238}U = \frac{\ell n2 \times 6.022 \times 10^{23}}{4.4683 \times 10^{9} \times 5.2596 \times 10^{5} \times 2.3805 \times 10^{2}}$$

$$= 7.461 \times 10^{5} \text{ dis/min-gram}$$

$$= 3.361 \times 10^{-1} \text{ pCi/}\mu g \times 99.284\% = 0.3337 \text{ pCi/g of normal U}$$

Therefore, the activity of 1 μg of normal uranium is

0.331 pCi ^{234}U + 0.0154 pCi ^{235}U + 0.3337 pCi ^{238}U

= 0.680 pCi/μg of normal U.

Note that 2.26% of the total activity is due to ^{235}U and, therefore, 48.87% each is due to ^{234}U and ^{238}U.

Example Calculation: 4-S10-A

3.4 ± 0.4 μg/gram x 0.68 pCi/μg = 2.3 ± 0.3 pCi/g.

APPENDIX 6

PERTINENT RADIOLOGICAL REGULATIONS STANDARDS, AND GUIDELINES

I. Excerpts From

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.

TABLE 1

SURFACE CONTAMINATION LIMITS*

Contaminants			Limit (Activity) (dis/min-100 cm^2)+	
Group	Description	Nuclides (Note 1)	Removable	Total (Fixed plus Removable)
1	Nuclides for which the nonoccupational MPC_a (Note 2) is 2×10^{-13} 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 or 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 $1/n \sum_{n} S_i \geqq L$, 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) 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 $\leqq$ 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.)

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 $1/n \sum_n S_i \geqq L$, 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 $\leqq$ 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. 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, JULY 1982

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

The instructions in this guide, in conjunction with Table 1, specify the radionuclides and radiation exposure rate limits which should be used in 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 the application of 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.

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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 Administrator of the NRC Regional Office 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.

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TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, ^{235}U, ^{238}U and associated decay products	5,000 dis/min-100 cm^2 α	15,000 dis/min-100 cm^2 α	1,000 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	1,000 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.	5,000 dis/min-100 cm^2 $\beta\gamma$	15,000 dis/min-100 cm^2 $\beta\gamma$	1,000 dis/min-100 cm^2 $\beta\gamma$

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TABLE 1
(Footnotes)

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.

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

712.1 Purpose

(a) The regulations in this part establish the criteria 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.

(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.

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(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.

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

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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 of 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.

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

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(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 action exists.

(C) If the external gamma radiation level is equal to or greater than 0.001 mR/h above backgrond 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 occuping affected area, occupancy time, and use of the affected area.

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

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

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

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

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

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

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

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

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

712.10 Selection of appropriate remedial action.

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

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

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

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

192.00 Applicability

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

192.01 Definitions

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

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

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

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

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

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

192.02 Standards

Control shall be designed* to:

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

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

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

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

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

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

192.10 Applicability

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

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

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

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

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

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

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

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

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192.11 Definitions

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

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

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

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

192.12 Standards

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

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

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

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

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

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

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

SUBPART C--Implementation

192.20 Guidance for Implementation

Section 108 of the Act requires the Secretary of Energy to select and perform remedial actions with the concurrence of the Nuclear Regulatory Commission and the full participation of any State that apys part of the cost,

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and in consultation, as appropriate, with affected Indian Tribes and the Secretary of the Interior. These parties, in their respective roles under Section 108, are referred to hereafter as "the implementing agencies."

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

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

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

If contaminants have been released from a tailings pile, an assessment of the location of the contaminants and the rate and direction of movement of contaminated groundwater, as well as its relative contamination, should be made. In addition, the assessment should identify the attenuative capacity of the unsaturated and saturated zone to determine the extent of plume movement. Judgments on the possible need for remedial or protective actions for groundwater aquifers should be guided by relevant considerations described in EPA's hazardous waste management system (47 FR 32274, July 26, 1982) and by relevant State and Federal Qater Quality Criteria for anticipated or existing uses of water over the term of the stabilization. The decision on whether to institute remedial action, what specific action to take, and to what levels an aquifer

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should be protected or restored should be made on a case-by-case basis taking into account such factors as technical feasibility of improving the aquifer in its hydrogeologic setting, the cost of applicable restorative or protective programs, the present and future value of the aquifer as a water resource, the availability of alternative water suppliues, and the degree to which human exposure is likely to occur.

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

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

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

192.21 Criteria for Applying Supplemental Standards

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

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

(b) Remedial actions to satisfy the cleanup standards for land, Section 192.12(a), or the acquisition of minimum materials required for control to satisfy Section 192.02(b), would, notwithstanding reasonable measures to limit damage, directly produce environmental harm that is clearly excessive compared

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to the health benefits to persons living on or near the site, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that may reasonably be anticipated.

(c) The estimated cost of remedial action to satisfy Section 192.12(a) at a "vicinity" site [described under Section 101(6)(B) of the Act] is unreasonably high relative to the long-term benefits, and the residual radioactive materials do not pose a clear present or future hazard. The likelihood that buildings will be erected or that people will spend long periods of time at such a vicinity site should be considered in evaluating this hazard. Remedial action will generally not be necessary where residual radioactive materials have been placed semi-permanently in a location where site-specific factors limit their hazard and from which they are costly or difficult to remove, or where only minor quantities of residual radioactive materials are invovled. Examples are residual radioactive materials under hard surface public roads and sidewalks, around public sewer lines, or in fence post foundations. Supplemental standards should not be applied at such sites, however, if individuals are likely to be exposed for long periods of time to radiation from such materials at levels above those that would prevail under Section 192.12(a).

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

(e) There is no known remedial action.

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

192.22 Supplemental Standards

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

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

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

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

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

"Interim Soil Limits for D&D Projects"

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

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

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

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

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

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

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VI. EXCERPTS FROM DOE 5480.1 Chg. 6, 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
External and Internal Exposure
of Members of the Public

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

APPENDIX 7

DOSE-DETERMINATION CALCULATION

The external penetrating radiation dose rates were measured through an ~ 7 mg/cm^2 end-window beta-gamma Geiger-Mueller (GM) detector. In addition to ambient background measurements, readings were obtained on contact and at 1 m from areas of contamination. A few of the end-window GM exposure readings at 1 m were distinguishable from the instrument background (0.02-0.03 mR/h). The highest (0.2 mR/h) was at location 41. It is assumed that the half-life of the contaminant is relatively long and the dose rate is, therefore, constant over a considerable time. Assuming that a person was stationary at this location for 40 hours per week for 50 weeks a year (a conservative assumption), the annual dose (A) for such an exposure is calculated as follows:

A = 40 hr/week x 50 weeks/year x (0.2 mR/h - 0.03 mR/h background)

= 2000 hrs/yr x 0.17 mR/h above background

= 340 mR/yr

= 340 mrem/yr

where, for the purpose of this report, a millirem of x- and gamma-ray radiation is considered equivalent to one milliRoentgen. The maximum, albeit unlikely, annual external penetrating radiation dose is therefore 340 mrem.

To assess the internal radiological hazard from inhalation/ingestion of contamination, a hypothetical, yet conceivable, worst-case situation involving the torching or welding of pipe in a tank was postulated. Based on the results of gamma-spectral analyses of samples of contamination from the tanks, normal uranium has been used as the nuclide(s) of contamination in the scenario that follows.

The highest level of contamination found in the tanks was 3.4 x 10^5 dis/min-100 cm^2 equated to normal uranium. This was found on the inlet pipe inside of filtered liquor tank-6 and on the crossover pipe inside of filtered liquor tank-5. For this scenario it has been assumed that it is necessary to weld a pipe at the location of one of the spots of contamination. The spots of contamination were no larger than 500 cm^2, therefore, the total activity (B) is:

$$B = \frac{3.4 \times 10^5 \text{ dis/min}}{100 \text{ cm}^2} \times 500 \text{ cm}^2 = 1.7 \times 10^6 \text{ dis/min}$$

APPENDIX 7
(cont'd.)

The activity (C) in units of μCi is:

$$C = 1.7 \times 10^6 \text{ dis/min} \times \frac{1 \text{ Ci of normal U}^*}{4.54 \times 10^{12} \text{ dis/min}} \times \frac{10^6 \text{ μCi}}{1 \text{ Ci}}$$

$$= 3.7 \times 10^{-1} \text{ μCi}$$

If this contaminated area was torched, a radioactive aerosol would be created. This aerosol would probably be nearly contained within the tank. The tank has a diameter of 4.6 m (15 ft) and is approximately 3.0 m (10 ft) in height. Thus, the total volume (D) of the tank is:

$$D = (\pi r^2)h = \pi(2.3 \text{ m})^2 \times 3.0 \text{ m} = 5.0 \times 10^1 \text{ m}^3$$

If 90% of the radioactivity becomes airborne and respirable, then the concentration of normal uranium (E) in the air in the tank would be:

$$E = 3.7 \times 10^{-1} \text{ μCi} \times 0.90/5.0 \times 10^1 \text{ m}^3$$

$$= 6.7 \times 10^{-3} \text{ μCi/m}^3$$

It is doubtful that more than one person would be involved in this situation for an extended length of time. Assuming a person would inhale 1.2 m^3 of air per hour (Ref. 1) and would be involved in this job for one hour, the amount of activity (F) that would be inhaled is:

$$F = 6.7 \times 10^{-3} \text{ μCi/m}^3 \times 1.2 \text{ m}^3/\text{h} \times 1 \text{ h}$$

$$= 8.0 \times 10^{-3} \text{ μCi}$$

$$= 8.0 \times 10^3 \text{ pCi}$$

The adult inhalation dose commitment factors for the bone, kidney, lung, and total body from ^{238}U, ^{234}U, ^{235}U, and short-lived daughters (Ref. 2) are presented in Table 7.1. The sum of the factors for ^{238}U and ^{234}U and short-lived daughters is also presented. Utilizing the results of the calculations given in Appendix 5, i.e., that 2.26% of normal uranium disintegrations per minute are due to ^{235}U and 97.747% due to $^{238,234}U$ (or 48.87% each), the dose commitment factors for normal uranium are obtained and are presented in terms of pCi of ^{238}U.

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*A curie of normal uranium normalized to ^{238}U, i.e., the sum of 3.7×10^{10} dis/s from ^{238}U, plus 3.7×10^{10} dis/s from ^{234}U, plus 1.7×10^9 dis/s from ^{235}U. This equals 7.57×10^{10} dis/s or 4.54×10^{12} dis/min. A standard curie is 3.7×10^{10} dis/s or 2.22×10^{12} dis/min.

APPENDIX 7
(cont'd.)

The 50-year dose commitment (G) from the inhalation of 8.0×10^3 pCi of normal uranium is:

$G = 8.0 \times 10^3$ pCi x

(1) 1.04 mrem/pCi inhaled = 8.3×10^3 mrem, lung

(2) 1.53×10^{-2} mrem/pCi inhaled = 1.2×10^2 mrem, bone

(3) 3.27×10^{-3} mrem/pCi inhaled = 2.6×10^1 mrem, kidney

(4) 3.17×10^{-2} mrem/pCi inhaled = 2.5×10^2 mrem, total body

Thus, the person would receive an 8300-mrem dose commitment to the lung, an 120-mrem dose commitment to the bone, a 26-mrem dose commitment to the kidneys, and a 250-mrem dose commitment to the total body under this scenario.

Even though these calculations are based on reasonable hypothesized values, the actual total activity inhaled and subsequent dose commitments could differ from that hypothesized. This is due to uncertainties in the estimation of activity in the steel pipe, in the estimation of the fraction that becomes airborne and respirable, in the estimation of the breathing rate and duration of inhalation and in the application of the dose commitment factors to the person involved. The hypothesized case is, however, based on reasonably conservative assumptions and, therefore, most probably overestimates the true potential situation.

REFERENCES

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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.

APPENDIX 7
(cont'd.)

TABLE 7.1

ADULT DOSE-COMMITMENT FACTORS
(mrem/50 yr per pCi inhaled in the 1st year)

Nuclide	Bone	Kidney	Lung	Total Body
^{238}U	7.1×10^{-3}	1.5×10^{-3}	4.80×10^{-1}	1.5×10^{-2}
^{234}Th	9.67×10^{-7}	5.73×10^{-8}	2.4×10^{-4}	3.91×10^{-6}
^{234m}Pa	2.97×10^{-12}	2.36×10^{-12}	3.15×10^{-8}	4.53×10^{-10}
^{234}U	7.9×10^{-3}	1.7×10^{-3}	3.36×10^{-1}	1.6×10^{-2}
^{234}U & ^{238}U & short-lived daughters (per pCi of ^{238}U)	1.50×10^{-2}	3.20×10^{-3}	1.016	3.10×10^{-2}
^{235}U	7.20×10^{-3}	1.50×10^{-3}	4.84×10^{-1}	1.50×10^{-2}
^{231}Th	8.72×10^{-8}	1.64×10^{-8}	4.16×10^{-6}	1.01×10^{-7}
^{235}U & ^{231}Th	7.20×10^{-3}	1.50×10^{-3}	4.84×10^{-1}	1.50×10^{-2}
normal U*	7.49×10^{-3}	1.60×10^{-3}	5.08×10^{-1}	1.55×10^{-2}
normal U* (per pCi of ^{238}U)	1.53×10^{-2}	3.27×10^{-3}	1.04	3.17×10^{-2}

*Normal U is 2.26% ^{235}U, 97.74% ^{234}U and ^{238}U, by pCi (see Appendix 5).

APPENDIX 8

EVALUATION OF RADIATION EXPOSURES AT BUILDING 55, CHEMICALS GROUP, OLIN CORPORATION, JOLIET, ILLINOIS

I. PREFACE

The U.S. Department of Energy has initiated a program to determine the present radiological condition of sites formerly used for work with radioactive material by the Manhattan Engineer District (MED) and the Atomic Energy Commission (AEC). In 1951, the Blockson Chemical Company entered into a contract with the AEC to conduct a development program for the extraction of uranium from wet-process phosphoric acid. The contract included the operation of a uranium-extraction facility in Building 55 at the firm's Joliet, Illinois, site. Production was ultimately limited to not more than 50,000 pounds of uranium annually. In 1955, Blockson Chemical Company was sold to the Olin Mathieson Chemical Corporation, now known as Chemicals Group, Olin Corporation. The uranium-extraction work was completed in 1962 when the contract with the AEC expired. Since existing documentation was insufficient to determine whether any decontamination work done at the time nuclear activities ceased was adequate by current guidelines, a comprehensive radiological assessment of Building 55 was conducted on an intermittent basis from March 27 to November 28, 1978.

Building 55 is a 30.5-m (100 ft) by 53.3-m (175 ft) brick structure with four levels. It contains various processing tanks and mixing vats. At the time of the survey, Building 55 was still used for the chemical processing of phosphate products from ground phosphate rock of Florida origin.

II. 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 the 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 existance 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.

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(cont'd.)

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 not stable and will spontaneously emit radiation in order to achieve a more stable state. By spontaneously transforming itself, the ratio of protons and neutrons in the nucleus 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 occuring 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 to

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(cont'd.)

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 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 produces a significant portion of the whole-body radiation dose to man.

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.

III. 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.

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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 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 particle 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. 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.

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(cont'd.)

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

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

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

IV. 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

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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 resulted 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 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

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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 "Requirements for Radiation Protection," give limits for external and internal exposure 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 given limits.

V. RESULTS OF SITE RADIOLOGICAL SURVEY

The comprehensive radiological survey performed at Building 55 was conducted on an intermittent basis between March 27 and November 28, 1978. Direct instrument surveys and smear surveys indicated that some contamination is present throughout the building, mainly on the concrete floors, the overhead beams, and on the mixing vats and processing tanks. Gamma-spectral analyses indicated that the contaminant is predominantly normal uranium. Thirty-three spots on localized areas and three larger general areas within Building 55 exceeded the acceptable surface-contamination limits for uranium. Two spots or localized areas on the roof and the general roof area exceeded the acceptable limits for radium. In 15 instances within the building, the contamination was found to be easily removable when smeared. Air sampling inside the building indicated ranges of radon-daughter concentrations within the range of normally expected

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background concentrations. No long-lived radionuclides were detected in any air sample. Environmental soil sampling about the grounds of Building 55 indicated significantly elevated levels of uranium and radium at two sampling locations near the building.

The survey data may be evaluated in terms of the potential doses that exposed persons could receive. Doses were calculated for a scenario that would result in a presumed maximum external penetrating radiation dose and for a pathway that could result in the presumed maximum internal radiation dose from inhalation of radioactivity. The maximum potential external dose was calculated to be 340 mrem per year, which represents an increase of about 410% above the 82 mrem annual natural background whole-body dose and 68% of the 500-mrem limit for an individual of the public. The maximum potential internal 50-year dose commitment was calculated to be 8300 mrem to the lung, 120 mrem to the bone, 26 mrem to the kidney, and 250 mrem to the whole-body. For the lung, bone, and kidney, these represent about 4600%, 79%, and 32% of the 179-mrem, 171-mrem, and 82-mrem annual internal background lung, bone, and kidney doses, respectively, and 550%, 8%, and 1.7% of the 1500-mrem limit for an individual of the public. For the whole body, this is about 300% of the 82-mrem annual natural background whole body dose and 50% of the 500 mrem limit for an individual member of the public.

To reduce the potential for radiation exposure, remedial measures such as stabilization of the contamination in place would be applicable as a short-term measure. To reduce the risk in the event that building modifications take place in the future, health physics procedures and coverage are recommended. The long-term solution would involve decontamination by removal of the radioactive residues from the areas in the facility where contamination was detected.

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