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INTERIM REPORT

RADIOLOGICAL SURVEY OF THE PROPERTY AT 9200 LATTY AVENUE, HAZELWOOD, MISSOURI

R. W. Leggett
P. T. Perdue
F. F. Haywood
E. B. Wagner
M. T. Ryan
C. J. Barton

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R. W. Leggett F. F. Haywood M. T. Ryan
P. T. Perdue E. B. Wagner C. J. Barton

Health and Safety Research Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

ABSTRACT

A radiological survey was conducted at an 11-acre site in Hazelwood, Missouri, formerly used for storage and drying of radioactive residues originating from uranium ore handling operations at the Mallinckrodt Chemical Works in St. Louis, Missouri. There are four buildings on the site; at the time of the survey, these buildings were being prepared for use in an operation for manufacturing chemical coatings. The survey included measurement of the following: external gamma radiation at 1 m above surfaces throughout the site; beta-gamma dose rates at 1 cm from surfaces throughout the site, and transferable beta contamination levels in the buildings; directly measured and transferable alpha contamination levels on surfaces in the buildings; concentrations of ^{226}Ra, ^{238}U, ^{232}Th, ^{227}Ac, and/or ^{230}Th in samples of soil and other materials on the site; concentrations of ^{230}Th, ^{226}Ra, and ^{210}Pb in water and sediment from a creek near the site; and concentrations of radon and progeny in air in the buildings on the site. It was found that alpha and beta-gamma contamination levels on surfaces and external gamma radiation levels at 1 m above surfaces exceed current NRC guidelines for the release of

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decontaminated property for unrestricted use over a large portion of the site. It is estimated that the top three inches of soil on much of the site contains an average of 140 pCi $^{226}Ra/g$ and probably higher average concentrations of ^{238}U, ^{230}Th, and ^{227}Ac. If reconstruction of the buildings is completed without removal of waste residues presently inside or near the structures, potentially hazardous ^{222}Rn daughter concentrations could develop inside some of the buildings. At present, the concentration of ^{219}Rn daughters in Building 1 appears to be higher than the ^{222}Rn daughter concentration.

INTRODUCTION

At the request of the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC), a radiological survey was conducted at a former licensed site located at 9200 Latty Avenue in Hazelwood, Missouri. There are four buildings on this 11-acre site (see Figs. 1 and 2). These structures have a total floor area of approximately 18,000 ft^2. At the time of the survey, the buildings were all vacant and were being prepared by 4 workers for use in an operation for manufacturing chemical coatings; the remaining 10.6 acres on the property were not being used. The site is located in a large industrial area and is well removed from residential areas. Water run-off from some parts of the property drains into the nearby Coldwater Creek and the remainder of the run-off apparently drains into the city sewer system.

The following history of the site was obtained from a 1976 NRC report.[1] In early 1966, ore residues and uranium- and radium-bearing

processed wastes which had been stored at the St. Louis Airport property were moved by the Continental Mining and Milling Company of Chicago, Illinois, to the Latty Avenue site. These wastes had been generated by Mallinckrodt Chemical Works of St. Louis during the period 1942 through the late 1950's. In January, 1967, the Commercial Discount Corporation of Chicago, Illinois, purchased the residues; much of the material was then dried and shipped to the Cotter Corporation facilities in Canon City, Colorado. The source material remaining at the Latty Avenue site was sold to the Cotter Corporation in December, 1969. Records indicate that residues remaining on the site at that time included 74,000 tons of Belgian Congo pitchblende raffinate containing about 113 tons of uranium, 32,500 tons of Colorado raffinate containing about 48 tons of uranium, and 8700 tons of leached barium sulfate containing about 7 tons of uranium. During the period August through November, 1970, Cotter Corporation dried some of the remaining residues and shipped them to their mill in Canon City, Colorado; by December, 1970, an estimated 10,000 tons of Colorado raffinate and 8700 tons of leached barium sulfate remained at the Latty Avenue site.

In April 1974, an NRC inspector was informed that the remaining Colorado raffinate had been shipped in mid-1973 to Canon City without drying and that the leached barium sulfate had been transported to a landfill area in St. Louis County. A reported 12 to 18 in. of top soil had been stripped from the Latty Avenue site and had supposedly been removed with the leached barium sulfate. However, analyses of soil samples taken during an NRC investigation of the Latty Avenue site in

1976 indicated the presence of uranium- and thorium-bearing residues; furthermore, at some points on the site, direct readings of radiation exceeded criteria established by NRC for decontamination of land areas prior to release for unrestricted use.

The present survey was undertaken to characterize the existing radiological status of the property. It was conducted by five members of the Health and Safety Research Division, Oak Ridge National Laboratory (ORNL) during the periods June 27 through July 1, July 11 through July 15, and September 19 through September 22, 1977. The survey included:

(1) measurement of external gamma radiation levels at 1 m above the surface throughout each building, and outdoors throughout the site;

(2) measurement of fixed and transferable alpha and beta contamination levels on walls, ceilings, floors, and supports throughout each building;

(3) measurement of beta-gamma dose rates at 1 cm from surfaces throughout the buildings and outdoors throughout the site;

(4) measurement of ^{226}Ra, ^{238}U, ^{232}Th, and ^{227}Ac (and, in a few cases, ^{230}Th) in soil samples taken inside buildings and outdoors on the site;

(5) measurement of ^{230}Th, ^{226}Ra, and ^{210}Pb in water samples taken from Coldwater Creek near the site; and

(6) measurement of radon and progeny in air in the buildings on the site.

"Contamination," as used in this report, refers to radioactive materials either on or below surfaces, whether fixed or removable. Survey meter readings made on surfaces are used to estimate the level of total surface contamination, while standard smear techniques are used to estimate the levels of transferable contamination.

SURVEY METHODS

Throughout this report, the term "lower wall" refers to the surface of a wall up to a height of 6 ft, and "overhead surfaces" include wall surfaces above 6 ft. A "survey block" is a rectangular subsection of some large area to be surveyed, either indoors or outdoors. Survey blocks are normally formed by mutually perpendicular sets of "grid lines," and the intersection of these lines (that is, the corners of the survey blocks) are referred to as "grid points."

Floors and lower walls of the buildings on this site (except for Building 4) were divided into survey blocks using grid lines parallel to the bases of the walls. The survey blocks used in Building 1 measured 20 ft x 20 ft on the floor and 20 ft x 6 ft on the lower walls (see Figs. 3 through 6); those in Building 2 measured 20 ft x 17 ft on the floor and 20 ft x 6 ft on the lower walls (see Figs. 7 through 10); and those in Building 3 measured 7 ft x 7 ft on the floor and 7 ft x 6 ft on the lower walls (see Figs. 11 and 12). Also, the outdoor area on the site was divided into 50 ft x 50 ft survey blocks using two sets of grid lines, one parallel and the other perpendicular to a base line which had been run by a surveying company (see Figs. 1 and 2). Readings in Building 4 were taken at uniformly spaced, randomly selected points (see Figs. 13 and 14).

Measurement of Alpha Contamination Levels in Buildings

Direct measurements of alpha contamination levels were made on floor, wall, and ceiling surfaces throughout the buildings (except for dirt floors in Buildings 1 and 2) using alpha scintillation survey meters described in Appendix I. Grid point readings and maximum observed readings for survey blocks were recorded; and for areas not divided into survey blocks, readings were taken and recorded at nearly uniformly spaced points. The alpha survey meters were equipped with scalers which allow integration of counts over various time intervals, permitting direct alpha measurements on surfaces with low-level alpha contamination. If counts are integrated over a period of 60 sec, the count rate error* associated with a direct reading of 300 dpm/100 cm^2 is ± 30% for this instrument, and the count rate error associated with a direct reading of 100 dpm/100 cm^2 is ± 50%.

For the measurement of transferable alpha contamination levels, smear samples were taken, using standard smear techniques described in Appendix III, at many of the points at which direct alpha readings were taken. The smears were counted using the alpha smear counter shown in Appendix I. The count rate errors associated with transferable alpha measurements reported in this document are ± 30% or less. It should be pointed out that indeterminable errors are introduced in taking smear samples because of some variation in pressure applied, in the condition of the surface, and in surface area covered.

*Unless otherwise specified, errors reported in this document refer to the 68% confidence level.

Measurement of Beta-Gamma Dose Rates and Transferable Beta Contamination Levels in Buildings

Beta-gamma dose rates were measured at 1 cm from the floor, wall, and ceiling surfaces in the buildings, using Geiger-Muller survey meters described in Appendix I. In Buildings 1 and 2, readings were taken at each of the grid points on the floors and lower walls and at the point within each floor survey block showing the highest external gamma level at 1 m (as measured with the gamma scintillation survey meter described later). The survey blocks in Building 3 were considerably smaller, and each block was scanned with a G-M meter to determine the maximum beta-gamma dose rate. On all overhead surfaces in the buildings (including walls above 6 ft) and throughout Building 4, beta-gamma dose rates were measured at a sufficient number of uniformly spaced points to obtain representative data.

The G-M meters (with open-window probe) were calibrated at ORNL by comparison with a Victoreen Model 440 ionization chamber (see Appendix I), and a conversion factor of 2000 cpm = 1 mrad/hr was determined using materials contaminated with ^{226}Ra in approximate equilibrium with ^{238}U. At normal background levels, the G-M meter usually shows readings less than 0.05 mrad/hr (100 cpm) and averaging approximately 0.02 mrad/hr. It should be pointed out that readings within the range of normal background cannot be accurately reproduced on the G-M meter. For higher individual measurements it appears that, in extreme cases, the absolute error involved in using this conversion factor (2000 cpm = 1 mrad/hr) may be 60% or more. However, the absolute error involved in determining

an average beta-gamma dose rate for a large, contaminated surface, such as a floor or wall, appears to be no higher than 15%.

Beta radiation cannot penetrate the closed window on the G-M probe; hence, gamma radiation levels can be measured with the window closed. The conversion factor for gamma radiation is 3200 cpm = 1 mR/hr. This factor was determined at ORNL using a ^{226}Ra source. A significant difference in the open-window and closed-window readings on the G-M meter at some point indicates the presence of beta-emitting surface contamination, since most beta particles can penetrate only a few millimeters of dense materials.

The smear samples described earlier were counted, using beta smear counters shown in Appendix I, for the determination of transferable beta contamination levels. The error associated with measurement of transferable beta contamination is discussed in Appendix I.

Measurement of External Gamma Radiation Levels in the Buildings

Three types of instruments were used to measure external gamma radiation levels at 1 m above the surface on this site: (1) G-M meters described earlier; (2) a scaler-equipped G-M counter described in ref. 2; and (3) NaI scintillation meters described in Appendix I. The G-M meters are equipped with ratemeters which show instantaneous gamma radiation levels when the probe window is closed. Since these ratemeters are unreliable for radiation levels within the range of normal background, the scaler-equipped G-M counters were used to measure low-level gamma radiation. The NaI scintillation meters are extremely sensitive and were used on this site to locate contaminated areas, as well as to

locate points within survey blocks showing maximum external gamma levels. However, since the response of the NaI scintillation meter is highly energy dependent, actual measurements of external gamma radiation were made only with the two types of G-M instruments described earlier.

In Buildings 1 and 2, external gamma levels at 1 m above the floor were measured at each of the grid points. In addition, the maximum external gamma radiation level at 1 m was determined within alternate survey squares in Building 1 and in each survey square in Building 2. In Buildings 3 and 4, external gamma levels at 1 m were measured at randomly selected points, and in each of these buildings, a scan was made with the gamma scintillation meter to insure that the measurements had been representative for the building.

Measurement of Radon and Progeny in Air in the Buildings

Continuous 24-hr measurements of radon concentrations in air were made in Buildings 1, 2, 3, and 4 using an instrument developed by Wrenn, et al.,[3] and referred to as a Wrenn chamber. This instrument, described in Appendix II, was attached to a printer which recorded automatically the radon concentrations at intervals of 2000 sec. Because some radon and progeny from previous 2000-sec intervals remain in the Wrenn chamber, each reading actually represents a concentration which has, effectively, been integrated over a period of 2 to 4 hr.

For the measurement of radon daughter concentrations in air in Buildings 1 and 2, air was pumped for 5 to 10 min at approximately 12 liters per min through a membrane filter with a maximum pore size of 0.4 μm. The filter was counted using an alpha spectrometry technique described in Appendix II.

Measurements of External Gamma Radiation Levels at 1 m and Beta-Gamma Dose Rates at 1 cm from the Surface Outdoors

At each of the grid points shown in Fig. 1, the beta-gamma dose rate was measured with an open-window G-M meter at 1 cm from the surface, and the external gamma level was measured at 1 m above the surface with one of the G-M meters described earlier. Then each survey block was scanned with a NaI scintillation meter to determine the point showing the maximum external gamma level, and the beta-gamma dose rate was measured at 1 cm above that point with a G-M meter.

Determination of Radionuclide Concentrations in Soil and Water

Surface soil samples were taken from the dirt floors in Buildings 1 and 2, from a small open dirt area in Building 3, and outdoors at intervals of 150 ft or less; locations are shown in Fig. 15. Most samples were taken at randomly chosen locations; however, some sampling locations were chosen because of elevated radiation levels at those points. Core holes were drilled with a manually operated split-spoon sampler at indoor and outdoor locations (usually chosen at random) shown in Fig. 16, and soil samples were taken from these core holes at depths of 0 to 2 ft. In addition, samples of scale and building materials were taken from the interior surfaces of the buildings. All samples of soil and building materials were dried for 24 hr at 110°C and then pulverized to a particle size no greater than 500 μm in diameter (-35 mesh). Aliquots from each sample were transferred to plastic bottles, weighed, and counted using a Ge(Li) detector and a multichannel analyzer. The spectra obtained were analyzed by computer techniques. A description of

the Ge(Li) detector and soil counting techniques is given in Appendix III. Concentrations of ^{226}Ra, ^{238}U, ^{232}Th, and ^{227}Ac in the samples were determined.

A barrel containing a radioactive residue was found in the southwest corner of the property. The barrel was taken to ORNL, where samples of the residue were analyzed for ^{230}Th, ^{228}Th, ^{232}Th, ^{231}Pa, ^{227}Ac, and ^{226}Ra by the Analytical Chemistry Division using radiochemical techniques described in appendices to the ORNL Master Manual.

Water samples were taken at a point on the southern end of the site where drainage from the site flows into the nearby Coldwater Creek, and at a point downstream about 2 miles from this point. A third sample was taken on the northern end of the site at the point near Latty Avenue where drainage flows into a storm sewer. The water and sediment were analyzed for ^{230}Th, ^{226}Ra, and ^{210}Pb by the Analytical Chemistry Division of ORNL, using the radiochemical techniques mentioned above.

Background Measurements

Background external gamma radiation levels at 1 m above the ground were measured at 4 points within 4 miles of the site. The measurements ranged from 7 to 9 μR/hr. Soil samples taken at the same points contained, on the average, 1.3 pCi ^{226}Ra/g and 1.2 pCi ^{232}Th/g. The average concentration of ^{238}U in these samples was 1.2 pCi/g.

Background beta-gamma dose rates, as measured with the G-M meters used on the site, typically average approximately 0.02 mrad/hr. However,

as noted earlier, readings at this level cannot be accurately reproduced on the G-M meter. Background direct alpha readings for the type of alpha meter used on this site are negligible.

All direct meter readings reported in this document represent gross readings; background radiation levels have not been subtracted. Similarly, background levels have not been subtracted from radionuclide concentrations measured in environmental samples and building materials. For the measurement of transferable alpha and beta contamination levels, average background counts were determined for the smear counters (at the place of counting), and these background counts were subtracted from gross counts.

GUIDELINES USED TO EVALUATE RESULTS

Guidelines used in this document to evaluate data from the survey are provided in Appendix IV. Some of these guidelines are discussed briefly in this section.

Surface contamination levels measured on the site are compared in this report with NRC guidelines for release of property for unrestricted use. For surfaces contaminated with alpha emitters, strictest limits apply to a group of radionuclides including ^{226}Ra and ^{230}Th, two of the principal contaminants on this site. The average and maximum* limits for direct measurements of alpha contamination levels on surfaces contaminated with these radionuclides are 100 dpm/100 cm^2 and 300 dpm/100 cm^2, respectively, and transferable alpha contamination should not exceed 20 dpm/100 cm^2. Strictest NRC guidelines for beta emitters apply to a group of radionuclides including ^{227}Ac, which was also found to be in

*Measurements may not be brought below limits by averaging over more than 1 m^2. The maximum level applies to an area of not more than 100 cm^2.

abundance on this site. Transferable beta contamination levels for surfaces contaminated with ^{227}Ac should not exceed 20 dpm/100 cm^2, and, regardless of the contaminant, average and maximum beta-gamma dose rates should not exceed 0.20 mrad/hr and 1.0 mrad/hr, respectively.

There are several guidelines for external gamma radiation given in Appendix IV. One is the figure 500 mrem/year for maximum exposure to an individual in the population, recommended by the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection (NCRP), and stated in 10 CRF 20. Another is the Environmental Protection Agency's (EPA) recently adopted limit of 25 mrem/year for all phases of the uranium fuel cycle. Finally, there is the Surgeon General's figure of 50 μR/hr for exposure over the background level, above which remedial action was suggested in the Grand Junction, Colorado, situation. Since this site is being prepared for use as a small industry, it seems reasonable to assume that no individual will be on the site more than 2500 hr each year. Assuming an exposure time of 2500 hr per year, a gamma radiation level of more than 18 μR/hr (10 μR/hr plus the average background level of 8 μR/hr) would exceed the EPA recommendation, and a level of more than 208 μR/hr would exceed the ICRP recommendation. Furthermore, an external gamma level of more than 58 μR/hr would exceed the Surgeon General's guidelines.

SURVEY RESULTS

Measurement of Alpha and Beta-Gamma Contamination and and External Gamma Radiation Levels in Buildings

Building 1

This structure measures 120 ft x 100 ft and has a 30-ft ceiling. At the time of this survey, the floor of the building was composed of

dirt and broken concrete, and there were openings along the walls (including spaces for 33 windows) totaling approximately 2500 ft^2. This building was used at one time for drying and otherwise preparing radioactive residues for shipment.

Results of beta-gamma dose rate measurements in Building 1 are given in Figs. 3 and 4, and in Table 1. Highest contamination levels were measured on the dirt floor and on a horizontal steel ledge approximately 4.5 ft above the floor. Beta-gamma dose rates at 1 cm from the surface exceeded 0.20 mrad/hr over most points of measurement on the floor, wall, and ceiling surfaces and were as high as 2.4 mrad/hr at 1 cm above the dirt floor. The percentage of the dose rates attributable to beta radiation varied between zero and 80 on the floor and lower walls but was near zero at most points on overhead surfaces. On the lower walls, transferable beta contamination levels averaged 45 dpm/100 cm^2 and were as high as 250 dpm/100 cm^2 on the horizontal steel ledge mentioned above (see Table 2). On overhead surfaces, transferable beta contamination levels averaged 15 dpm/100 cm^2 but were as high as 200 dpm/100 cm^2 (Table 2). A general note should be made for surfaces where transferable beta contamination was measured. Contamination on a smear sample may not represent quantitatively the transferable component on the surface. Many smears were taken on surfaces where dust had accumulated to a thickness of 0.5 to 1 cm.

Direct alpha measurements on wall and ceiling surfaces in Building 1 generally exceeded 300 dpm/100 cm^2 (see Fig. 4 and Table 1) and were as high as 18,000 dpm/100 cm^2 on the steel ledge on the lower wall.

Direct alpha readings were taken at approximately 5 cm above the dirt floor at a few points; these readings exceeded 5,000 dpm/100 cm^2 at some points and probably resulted from the emanation of ^{222}Rn, ^{219}Rn, and possibly ^{220}Rn. Transferable alpha contamination levels on the walls and ceiling were generally higher than transferable beta levels; transferable alpha contamination levels averaged 115 dpm/100 cm^2 on the lower walls and 55 dpm/ 100 cm^2 on overhead surfaces (Table 2).

External gamma radiation levels at 1 m above the floor of Building 1 were generally in the range of 100 to 500 μR/hr (see Figs. 5 and 6). The average external gamma level at 1 m (based on grid point measurements) was approximately 190 μR/hr.

Building 2

This structure measures 60 ft x 50 ft and has a 25- to 30-ft ceiling and a dirt and gravel floor. At the time of the survey, the building had uncovered door, wall, and window openings totaling approximately 500 ft^2. The former use of this building is uncertain.

Results for beta-gamma dose rates measured in Building 2 are presented in Figs. 7 and 8, and Table 1. Beta-gamma dose rates at 1 cm from the surface were generally lower than in Building 1 but exceeded 0.20 mrad/hr in some areas of 1 m^2 or more on the floor. The fraction of the beta-gamma dose rate attributable to beta radiation was near zero at most points but was as high as 80% on the floor and 60% on the upper surfaces. Transferable beta contamination levels exceeded 20 dpm/100 cm^2 at several points on the lower walls and at a few overhead points (Table 2). The maximum transferable beta contamination level measured was 110 dpm/100 cm^2, on a steel ledge on the lower wall.

Direct measurements of alpha contamination levels on the walls and ceilings exceeded 500 dpm/100 cm^2 in several places and were as high as 2600 dpm/100 cm^2 (see Fig. 8 and Table 1). Maximum alpha contamination levels were generally measured on horizontal steel beams. Transferable alpha contamination levels exceeded 20 dpm/100 cm^2 at most points of measurement on the lower walls and at some points on overhead surfaces. The maximum transferable alpha level measured in the Building was 210 dpm/100 cm^2 (see Table 2); this smear sample was taken on a steel beam on an upper wall.

External gamma measurements at 1 m above the floor (Figs. 9 and 10) generally ranged from 40 to 105 μR/hr. It appeared that high gamma radiation levels outside the building were responsible for part (but not all) of the elevated beta-gamma dose rates and external gamma radiation levels inside the structure.

Building 3

This structure measures 42 ft x 28 ft and has a 15- to 20-ft ceiling and a concrete floor. It is referred to in an NRC report[1] as a "garage."

Beta-gamma dose rate measurements for Building 3 are reported in Fig. 11 and Table 1. At 1 cm from interior surfaces, beta-gamma dose rates did not exceed 0.18 mrad/hr except in the open dirt on the floor in the northwest corner of the building, where a maximum measurement of 0.40 mrad/hr (principally beta radiation) was recorded. Transferable beta radiation levels averaged 30 dpm/100 cm^2 on the floor and 20 dpm/100 cm^2 on the lower walls (Table 2).

Direct measurements of alpha contamination levels exceeded 500 dpm/100 cm^2 at points in each of the survey blocks on the floor and lower walls (see Fig. 11). On overhead surfaces, direct alpha measurements averaged 50 dpm/100 cm^2 (Table 1), and only 1 of 16 overhead measurements exceeded 500 dpm/100 cm^2; this reading was taken on a horizontal steel beam above the northern entrance. Transferable alpha contamination levels averaged 95 dpm/100 cm^2 on the floor and 50 dpm/100 cm^2 on the lower walls (Table 2). However, smear samples taken on vertical surfaces of the walls, (that is, not on beams and ledges) generally showed less than 20 alpha dpm/100 cm^2.

External gamma radiation levels measured at 1 m above the floor ranged from 50 to 55 μR/hr (see Fig. 12). These elevated readings are probably attributable chiefly (but not completely) to contamination outside the building.

Building 4

This small structure (56 ft x 20 ft) was partially destroyed in a fire and was undergoing extensive construction modifications at the time of the survey, particularly on the walls and ceiling. The building has a concrete floor. According to an NRC report,[1] this structure once served as an office building.

Radiation levels in Building 4 were generally low except for alpha contamination on the concrete floor. Direct alpha measurements on the floor were in the range of 50 to 530 dpm/100 cm^2 (see Fig. 13), and transferable alpha contamination levels were as high as 60 dpm/100 cm^2 (Table 2). It appeared that the elevated alpha readings were being

produced at least in part by soil brought into the building on shoes. External gamma measurements at 1 m were in the range 11 to 16 μR/hr (Fig. 14).

Measurements of Beta-Gamma Dose Rates at 1 cm and External Gamma Radiation Levels at 1 m Above Surfaces Outdoors

Measurements of beta-gamma dose rates at 1 cm above the grid points outdoors are presented in Fig. 17. The survey blocks showing beta-gamma dose rates greater than 0.20 mrad/hr at some point are shown in Fig. 18, along with the maximum observed dose rate for each of these blocks. Beta-gamma dose rates ranged from 0.20 mrad/hr to 5.0 mrad/hr over an estimated 30% of the surface area outdoors on the property. Based on measurements at grid points, the average beta-gamma dose rate outdoors on the site at 1 cm above the ground was approximately 0.25 mrad/hr.

External gamma measurements at 1 m above grid points outdoors on the site are given in Fig. 19. A radiation intensity profile for the outdoor area on the site is shown in Fig. 20. This profile reflects only average external gamma radiation levels at 1 m for the areas indicated and should not be interpreted as showing point-by-point radiation levels.

External gamma radiation levels at 1 m outdoors were as high as 500 μR/hr and exceeded 50 μR/hr over most of the property. The average external gamma radiation level at 1 m on the entire property, based on all grid-point measurements, was approximately 110 μR/hr.

Results of Soil Sample Analyses

Surface and subsurface samples taken from the floor of Building 1 (see Tables 3 and 4 and Figs. 15 and 16) contained up to 530 pCi $^{226}Ra/g$, up to 860 pCi $^{238}U/g$, and up to 700 pCi $^{227}Ac/g$. It appears from the corings that the contaminated soil in Building 1 extends to a depth of at least 18 in. in some places; however, the radionuclide concentrations below 6 in. at most points appear to be significantly lower than those from 0 to 6 in. Hole C13, which was drilled at a point near the wall where the dirt floor was higher than the nearby area, showed nearly uniform ^{226}Ra contamination (from 140 to 240 pCi/ $^{226}Ra/g$) from 0 to 18 in., and there was an interface between contaminated soil and relatively clean soil at approximately 18 in. It is estimated from the results in Tables 3 and 4 that the top 6 in. of soil on the floor in Building 1 has an average ^{226}Ra concentration of approximately 200 pCi/g and that the soil from 6 in. to 2 ft below the surface has an average ^{226}Ra concentration of 50 pCi/g or less.

Samples were taken at depths of 0 to 2 ft from two core holes in Building 2, and three surface samples were taken in this building (see Tables 2 and 4, and Figs. 15 and 16). Location S37 (indicated in Fig. 15) showed higher beta-gamma contamination levels than the rest of the floor, and a surface sample taken there showed 80 pCi $^{226}Ra/g$, 84 pCi $^{238}U/g$, and 47 pCi $^{227}Ac/g$. It is estimated from surface and core hole samples that the top 6 in. of soil in Building 2 contains an average ^{226}Ra concentration of approximately 30 pCi/g. Little contamination was found below 6 in. in the two core holes (see Table 4). A sample

of dirt and scale taken from a ledge near the ceiling in Building 2 contained 550 pCi ^{226}Ra/g, 566 pCi ^{238}U/g, 390 pCi ^{227}Ac/g, and 7 pCi ^{232}Th/g.

A surface soil sample taken from the open dirt on the floor in the northwest corner of Building 3 (S38) showed 82 pCi ^{226}Ra/g, 3.1 pCi ^{238}U/g, and 120 pCi ^{227}Ac/g. A sample of dirt and scale taken from a ledge in this building showed 80 pCi ^{226}Ra/g.

Soil sample analyses for outdoor surface samples are reported in Table 3; locations are shown in Fig. 15. Unsurprisingly, the concentration of ^{226}Ra in surface samples follows the same general pattern as that for external gamma radiation shown in Fig. 20. The sampling locations S7 through S34 (among others) were chosen independently of radiation levels, and average conditions at these 28 locations should be representative of average conditions for the entire site. Radium-226 concentrations in surface samples from these points averaged approximately 140 pCi ^{226}Ra. (The average external gamma level at 1 m at the same grid points was 125 μR/hr, as compared with an average of 110 μR/hr for all outdoor grid points on the site.) Concentrations of ^{226}Ra up to 2700 pCi/g, ^{238}U up to 210,000 pCi/g, and ^{227}Ac up to 1300 pCi/g were found in outdoor surface samples. The sample containing 210,000 pCi ^{238}U/g (~60% ^{238}U by weight) was composed of several pieces of yellow material found at location S49 shown in Fig. 15. It is interesting to note that dirt scraped from the boots of a surveyor who had walked in the area shown in Fig. 2 (including inside the buildings) showed 120 pCi ^{226}Ra/g, which is close to the estimated average for topsoil on the site. This sample also showed 110 pCi ^{227}Ac/g (see Table 3, sample S48).

Holes were cored to a depth of 2 ft or less at the outdoor locations shown in Fig. 16. Analyses of samples taken from these core holes are reported in Table 4. As in Buildings 1 and 2, the contaminated soil outdoors appears to be near the surface in most places, although contamination extends to a depth of at least 2 ft at some points. In particular, at location C5 (Fig. 16), the concentrations of ^{226}Ra, ^{238}U, and ^{227}Ac at a depth of 18 to 24 in. were 49 pCi/g, 150 pCi/g, and 37 pCi/g, respectively. Core hole C5 was located in an area which has been cultivated in recent years, and any contaminated materials which may have been on the surface are probably now distributed in the top foot or two of soil.

A barrel was found in the southwest corner of the property between grid points F19 and G19. A sample taken from the barrel and two surface samples taken on the site were analyzed for ^{230}Th, ^{228}Th, ^{232}Th, ^{231}Pa, ^{227}Ac, and ^{226}Ra by the Analytical Chemistry Division of ORNL, using radiochemical techniques. Results for these samples are reported in Table 5. The most abundant radionuclide found in the barrel sample was ^{230}Th; the concentration of this radionuclide was 90,000 pCi/g. The surface samples taken at locations S45 and S47 (Fig. 15) also showed high concentrations (18,000 and 36,000 pCi/g, respectively) of ^{230}Th.

Results of Water and Sediment Analyses

Results of analyses of water and sediment samples taken from a drainage path on the site and from Coldwater Creek near the site are reported in Table 6. The concentrations of ^{210}Pb, ^{226}Ra, and ^{230}Th in the water samples were well below the maximum permissible concentration

guides listed in 10 CFR 20, Appendix B, and in ERDA Manual Chapter 0524, Annex A. However, the presence of elevated concentrations of ^{210}Pb, ^{230}Th, and/or ^{226}Ra in sediment filtered from the two water samples taken from drainage at the boundary of the property indicates that some contamination is being transported from the property in water run-off.

Radon and Radon Daughter Measurements in Buildings 1, 2, 3, and 4

There are three radon isotopes produced in nature: ^{222}Rn (radon), ^{220}Rn (thoron), and ^{219}Rn (actinon). Radon-222 is in the ^{238}U (and ^{226}Ra) chain, ^{220}Rn is in the ^{232}Th chain, and ^{219}Rn is in the ^{235}U (and ^{227}Ac) chain. On sites where uranium-bearing materials have been processed or stored, the concentrations of ^{220}Rn and ^{219}Rn in air in buildings are usually negligible compared with concentrations of ^{222}Rn, and most methods used for measuring concentrations of ^{222}Rn and ^{222}Rn daughters are invalid, if significant concentrations of ^{220}Rn and ^{219}Rn are present. At the Latty Avenue site, not only are all three isotopes present, but the ^{219}Rn isotope appears to be dominant in Building 1 and possibly in Buildings 2 and 3. The mean life of ^{219}Rn is short (5.76 sec) compared to ^{222}Rn (5.5 days). Therefore ^{219}Rn which reaches the atmosphere in an air-over-contaminated ground gemoetry probably originates in the top millimeter of soil whereas ^{222}Rn in the air can originate from the maximum depth of contamination (in this case approximately 450 mm). Hence many more atoms of ^{222}Rn will escape from the soil than atoms of ^{219}Rn, and consequently, the concentration of ^{222}Rn in air may be 2 or 3 orders of magnitude higher than that of ^{219}Rn. On the other hand, the specific activity of ^{219}Rn is 8.4×10^4 greater than that for ^{222}Rn and probably accounts for the observed dominance of ^{219}Rn daughters.

Air samples were taken in all four buildings on this site. Filters used for collecting particulates were counted in an alpha counter and pulses were sorted and stored in one half of a 1024 channel pulse height analyzer (see Appendix II). The data corresponding to observed Radium A and Radium C' activity are evaluated using a technique refined by Kerr.[4] Generally, results are given for the concentrations of RaA, RaB, RaC, and working levels (WL); however, the presence of daughters of thoron or actinon creates interference and will yield erronenous estimates of these concentrations.

A typical radon daughter alpha spectrum is presented in Fig. 21. This spectrum was obtained by counting a sample taken from a radon source chamber used for instrument calibration. Radium A (6.05 MeV α) is seen to reside between channels 280 and 305. Likewise, the 7.69 MeV alpha for RaC' is found betweeen channels 375 and 395. In contrast to this spectrum is the one presented in Fig. 22, for a sample taken in Building 1. In this case it is seen that virtually all of the activity is due to Actinium C, which has a 6.28 MeV alpha residing between channels 295 and 312, and a 6.62 MeV alpha residing between channels 313 and 333. Similar spectra were observed in Buildings 2, 3, and 4 as seen in Figs. 23 through 25.

In Buildings 2 and 3, most of the activity may be attributed to actinium daughters. In Building 4, however, the predominate activity is due to radon daughters. During the period September 20 through 22, a total of 10 air samples were collected in Building 4 in order to evaluate the concentration of radon daughters. The maximum observed radon daughter concentration was 0.005 working levels and the average concentration was 0.003 pCi/liter. It appears from data in Figs. 22

through 25 that the concentration of thoron daughters is insignificant compared to daughters of radon and actinon. This is consistent with the fact that the quantity of ^{232}Th on the site is small compared with that of ^{226}Ra and ^{227}Ac.

The concentration of ^{211}Bi in Building 1 was estimated from alpha spectrometry to be as high as 10 pCi/liter. The fraction of equilibrium between ^{211}Bi and ^{219}Rn in the air sample showing this concentration is not known.

Measurements of ^{222}Rn concentrations in air in the buildings are reported in Table 7. These measurements were made with Wrenn chambers, which are covered with foam-rubber screens to filter radon daughters in air. The radon diffusion rate into the chamber is of the order of 30 min, which is long enough to remove the short-lived ^{219}Rn almost completely; the daughters of ^{219}Rn would be left in the foam rubber. Also, the contribution of ^{220}Rn to the Wrenn chamber measurements should be small, since there were only small quantities of this isotope present, and much of that present would be eliminated by decay in the foam-rubber cover. Consequently, it is thought that the accuracy of the ^{222}Rn measurements given in Table 7 was not decreased substantially by the presence of ^{220}Rn and ^{219}Rn in the buildings.

Maximum measured concentrations of ^{222}Rn in air in Buildings 1, 2, 3, and 4 were 57 pCi/liter, 7 pCi/liter, 1 pCi/liter, and 5.8 pCi/liter, respectively. Buildings 1, 2, and 3 are open, and it is expected that ^{222}Rn concentrations in these structures would be substantially higher

if the buildings were closed. Maximum ^{222}Rn concentrations in Buildings 1, 2, and 4 exceeded the maximum permissible concentration for nonoccupational exposure, 3.0 pCi/liter, stated in 10 CFR 20. Furthermore, since elevated ^{226}Ra concentrations were found in samples taken from ledges and open dirt in Building 3, it is not unlikely that the ^{222}Rn concentration in that structure would exceed 3.0 pCi/liter if the building were closed.

SUMMARY

Strictest NRC guidelines for the release of property for unrestricted use apply to surfaces contaminated with ^{226}Ra, ^{230}Th, and ^{227}Ac, among other nuclides. All three radionuclides were found to be in abundance on this site. It is estimated from soil analyses that the top 3 in. of soil outdoors on the property has an average ^{226}Ra concentration of approximately 140 pCi/g, and the activity of ^{227}Ac in many samples was as high or higher than the activity of ^{226}Ra. Two surface samples taken outdoors on the property showed ^{230}Th concentrations of 18,000 pCi/g and 36,000 pCi/g, and a sample of material taken from a barrel found on the property showed a ^{230}Th concentration of 90,000 pCi/g. One outdoor surface sample was approximately 60% ^{238}U by weight; this and other samples taken from the site contained quantities of ^{238}U exceeding the level (0.05% by weight) requiring a source material license.[5] It appears that most of the radioactive wastes outdoors and in Buildings 1 and 2 lie in the top 1 ft or less of soil, although some sampling points showed contamination at 18 in. or deeper.

The 120-ft x 100-ft building which had been used for drying of residues (Building 1) showed far higher contamination levels than the other three buildings on the site. This building has a dirt floor, and it is estimated that the top 6 in. of this dirt contains an average of 200 pCi ^{226}Ra/g. Beta-gamma dose rates at 1 cm from surfaces in Building 1 exceed the NRC guideline of 0.20 mrad/hr over a large fraction of the surfaces of this building, and direct alpha measurements generally far exceed the maximum NRC guidelines of 300 dpm/100 cm^2 for surfaces contaminated with ^{226}Ra or ^{230}Th. Furthermore, at many points in Building 1, transferable alpha contamination levels exceed the NRC guideline limit (for ^{226}Ra or ^{230}Th) of 20 dpm/100 cm^2, and transferable beta contamination levels exceed the limit (for ^{227}Ac) of 20 dpm/100 cm^2. Also, at some points in each of the other three buildings on the site, NRC guidelines for alpha and/or beta-gamma contamination are exceeded. In Building 2, a sample of dirt taken from the floor showed 80 pCi ^{226}Ra/g, and a sample of dirt and scale taken from an overhead beam showed 350 pCi ^{226}Ra/g and 566 pCi ^{238}U/g. It is estimated that the top 6 in. of floor dirt in Building 2 has a ^{226}Ra concentration of approximately 30 pCi/g. Building 3, formerly used as a garage, showed highest contamination levels in some open dirt in the northwest corner of the floor; a sample of this dirt showed a ^{226}Ra concentration of 82 pCi/g. Direct alpha readings in Building 3 exceed the NRC maximum guideline of 300 dpm/100 cm^2 at many points on the floor and lower walls. Direct alpha readings on the floor of Building 4 also exceed 300 dpm/100 cm^2 at some points;

it appears that the contamination in this building is due, at least in part, to dirt which has been brought in from outdoors on shoes.

Beta-gamma dose rates at 1 cm from the ground outdoors exceed the NRC guideline of 0.20 mrad/hr over an estimated 50% of the property. External gamma radiation levels at 1 m above the surface outdoors on the property averaged 110 μR/hr. Based on an exposure time of 2500 hr per year for an individual, the integrated exposure would be 275 mrem. This is in excess of the EPA recommended limit for an individual (25 mrem), but is approximately one half the limit recommended in 10 CFR 20 (500 mrem).

Concentrations of ^{222}Rn exceeding the limit of 3.0 pCi/liter stated in 10 CFR 20 were measured in Buildings 1, 2, and 4, even though Buildings 1 and 2 were open. This is not surprising in view of the large quantities of ^{226}Ra present in the soil inside and near the buildings. Since there are probably several air exchanges per hour in Buildings 1, 2, and 3, ^{222}Rn daughters attain only a small fraction of equilibrium with ^{222}Rn; consequently ^{222}Rn daughter concentrations are much lower than ^{222}Rn concentrations. However, if reconstruction of the buildings is completed with the residues left in place, it is likely that somewhat higher concentrations of ^{222}Rn and much higher concentrations of ^{222}Rn daughters than those now present will develop in Buildings 1 and 2 (and possibly in 3). Although it is difficult to predict the potential radon daughter concentrations in the (completed) buildings from the survey results, it should be pointed out that in a similar situation at the former Vitro

Rare Metals Plant in Canonsburg, Pennsylvania, ^{222}Rn concentrations exceeding 100 pCi/liter and ^{222}Rn daughter concentrations far exceeding the Surgeon General's guideline of 0.03 WL (see Appendix IV) were measured in industrial buildings built over or near radium-bearing wastes.[6]

The presence of large quantities of radionuclides in the ^{235}U chain (in particular, ^{227}Ac) has given rise to an unusual phenomenon--the occurrence of measurable quantities of airborne ^{219}Rn and daughters. In fact, it appears that the concentration of ^{219}Rn daughters is higher than that of ^{222}Rn daughters at some points on the site, particularly in Building 1. Little is known about the health hazards of ^{219}Rn and its daughters, or of the actinides that produce them. The limits stated in ERDA 0524 Annex A for protoactinium-231 and actinium-227 are comparable to those for the better known actinide, plutonium-239 (see Table 8). The principal dose from ^{219}Rn and daughters is to the lung, due to the short half-lives of the isotopes involved. Considering the half-lives and alpha energies of the daughters of ^{219}Rn and ^{222}Rn, it appears that the dose to the lung from ^{219}Rn daughters might be comparable to that from ^{222}Rn daughters; however, quantitative data on the relative hazard of these two radons is not presently available.

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4. G. D. Kerr, Measurement of Radon Progeny Concentrations in Air by Alpha-Particle Spectroscopy, ORNL-TM-4924, 1975.

5. Code of Federal Regulations, Title 10, Part 40, "Licensing of Source Materials."

6. R. W. Leggett, C. J. Barton, et al., Radiological Survey of the Former Vitro Rare Metals Plant, Canonsburg, Pennsylvania, Phase II, ERDA Report, 1977.

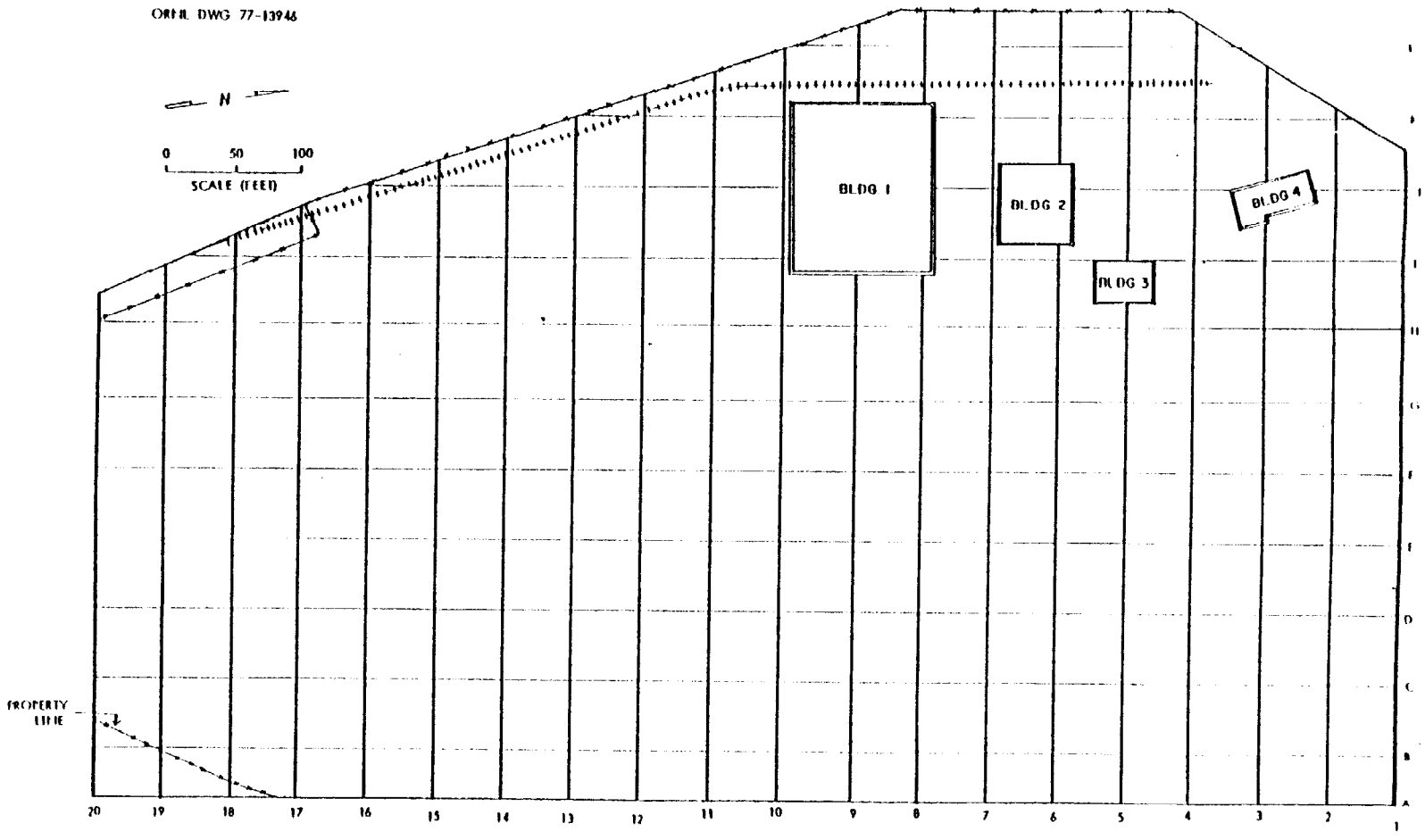

Fig. 1. Scaled drawing of the site and grid used for outdoor survey.

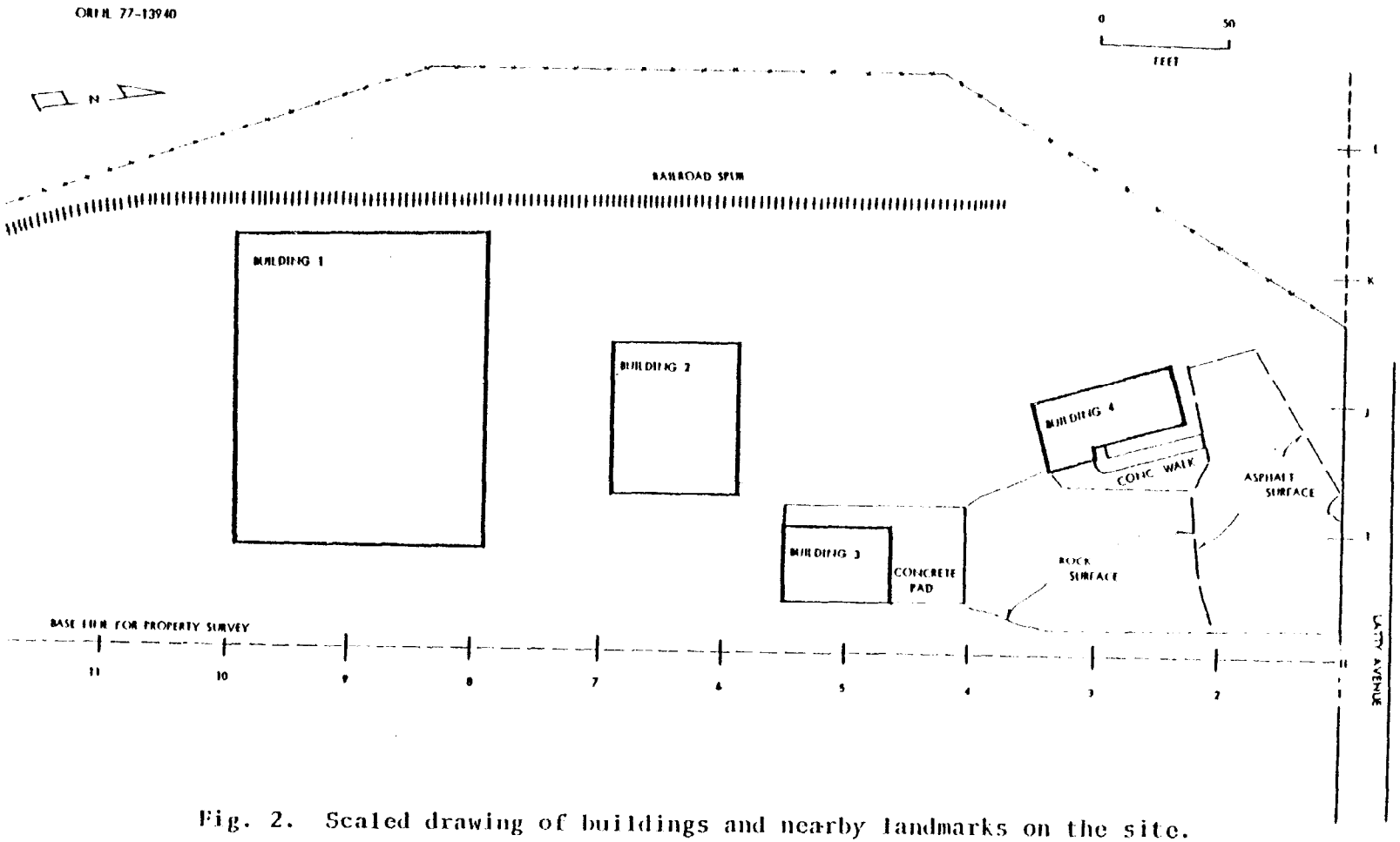

Fig. 2. Scaled drawing of buildings and nearby landmarks on the site.

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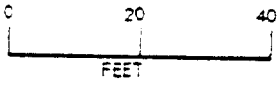

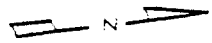

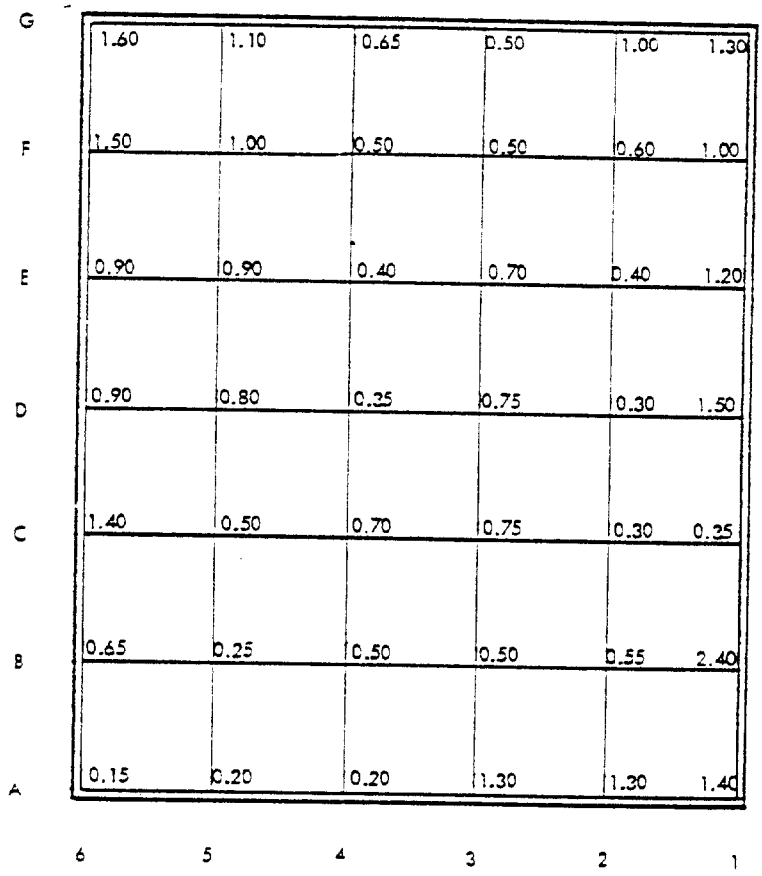

Fig. 3. Building 1, floor: beta-gamma dose rates (mrad/hr) at 1 cm above grid points.

ORNL 77-13935

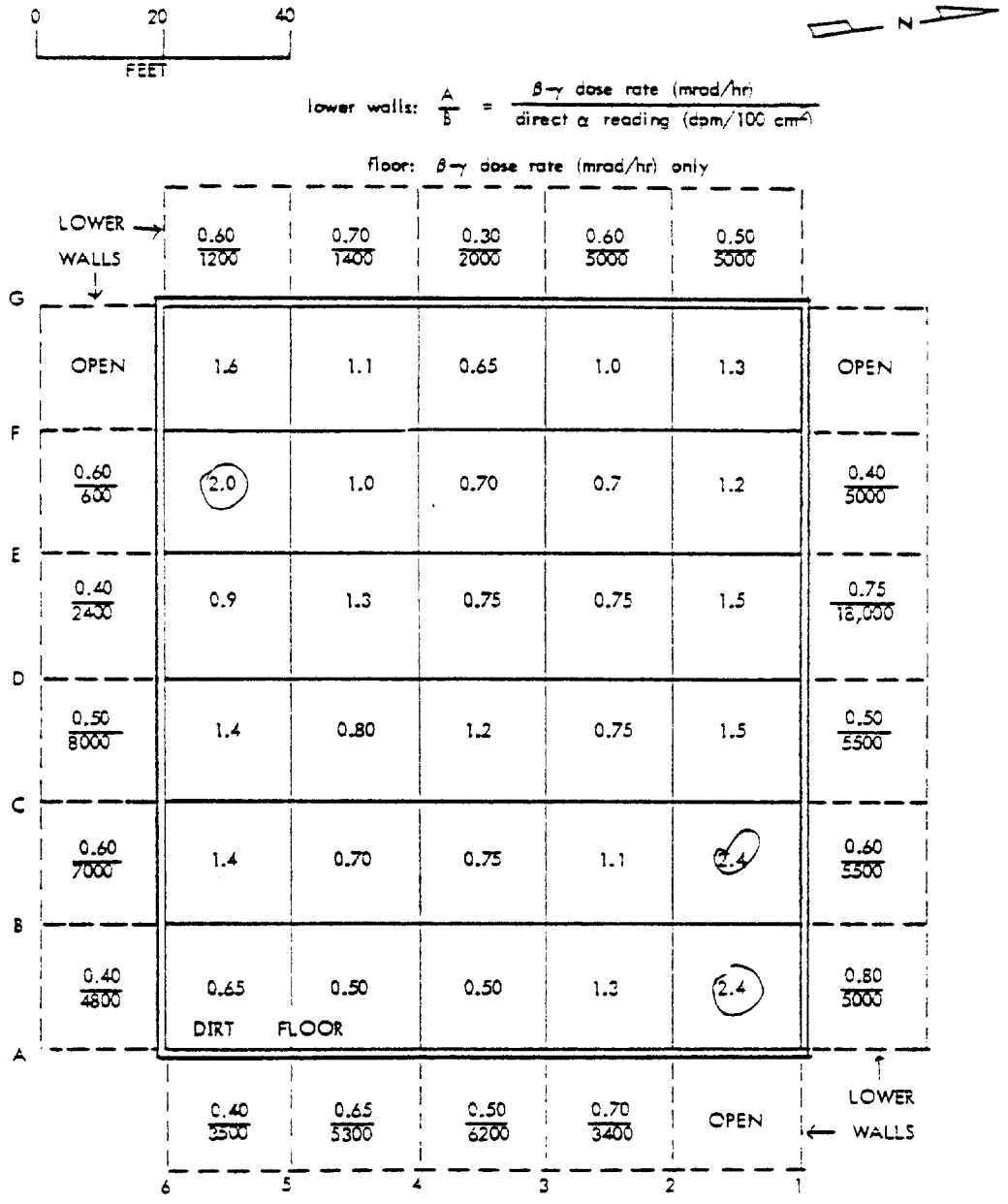

Fig. 4. Building 1, floor and lower walls: maximum observed beta-gamma dose rates and direct alpha readings.

ORNL DWG 77-16716

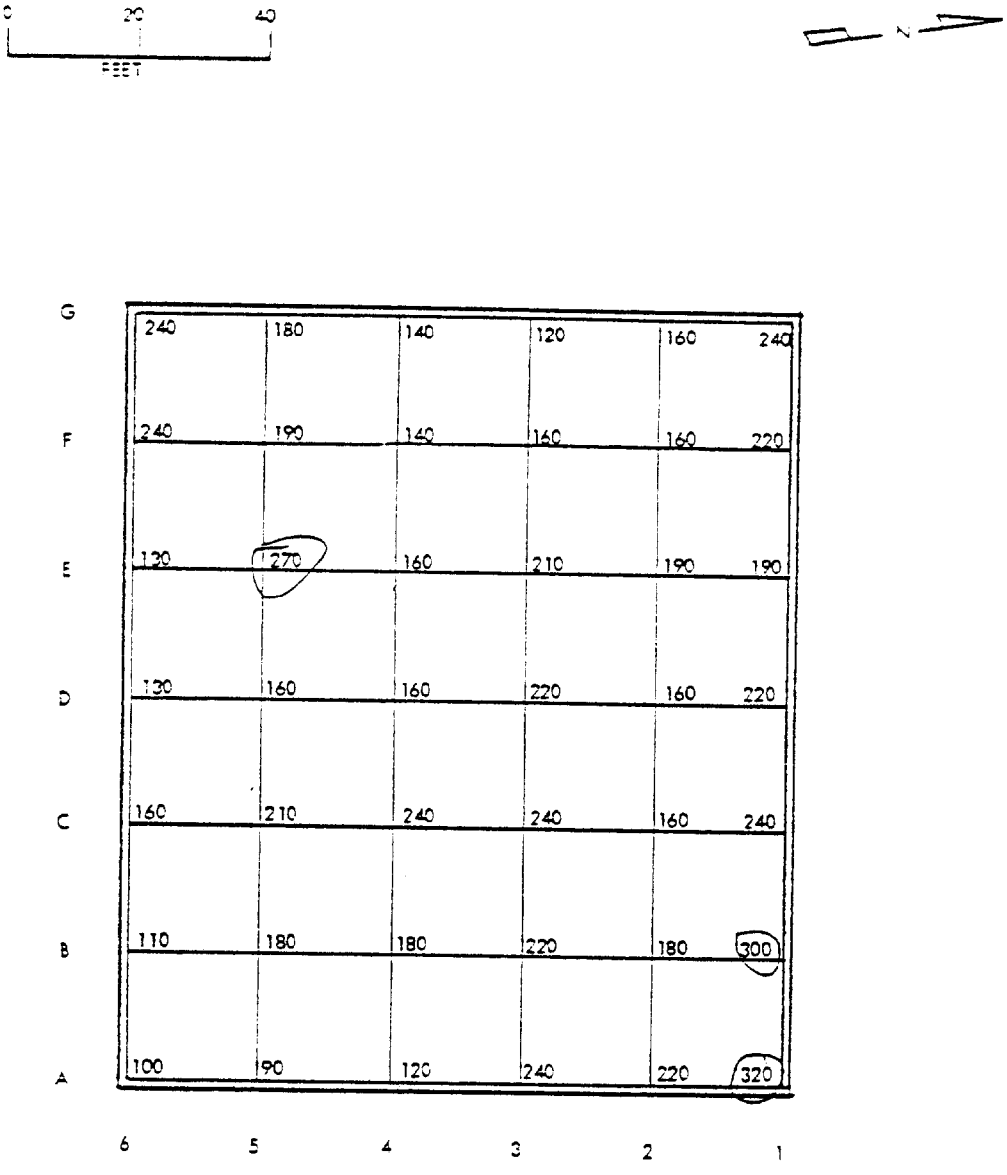

Fig. 5. Building 1, external gamma radiation levels (in μR/hr) at 1 m above grid points.

ORNL 77-13937

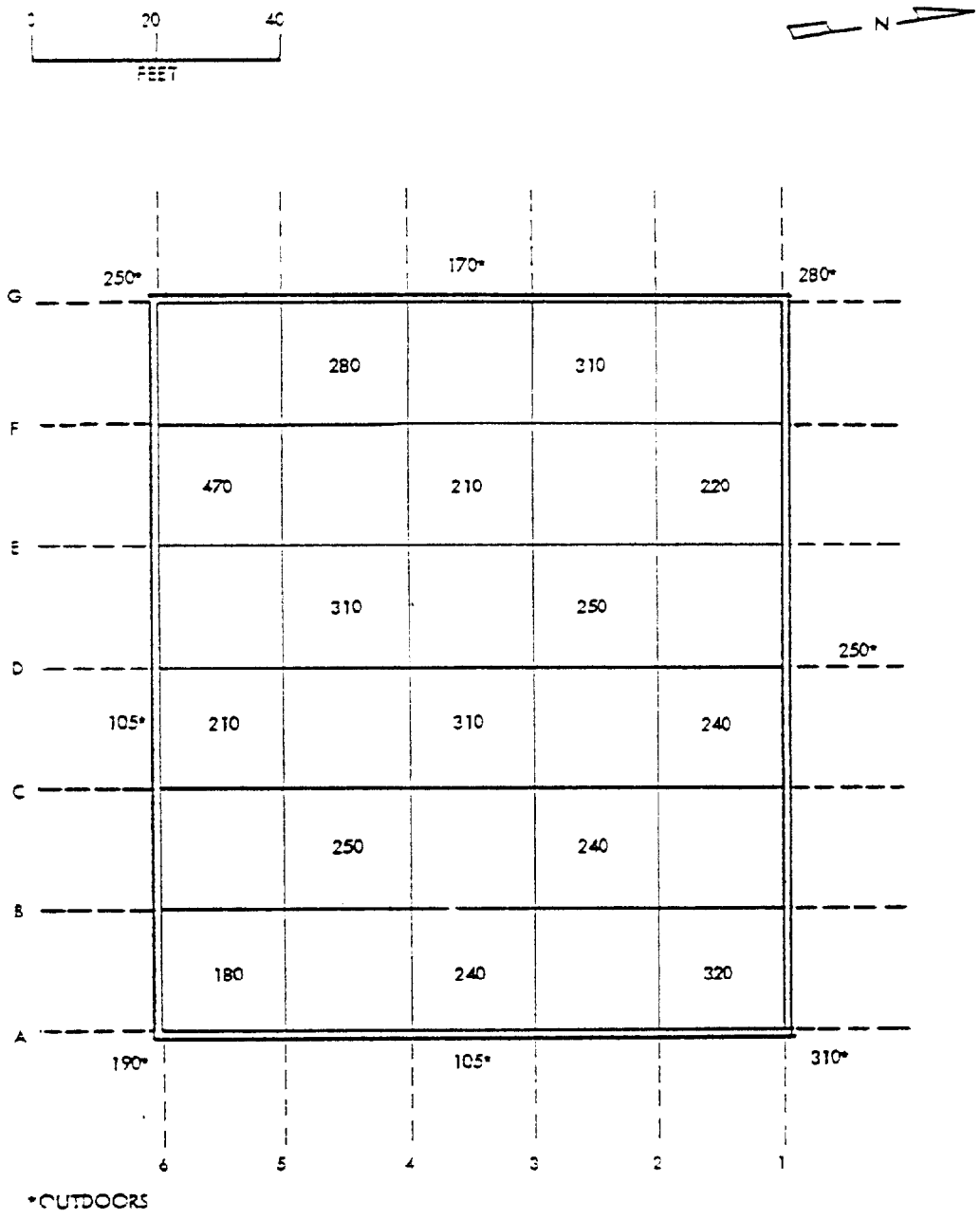

Fig. 6. Building 1; indoors and outdoors within 5 ft of walls; maximum external gamma radiation levels (in μR/hr) within survey blocks, at 1 m above floor.

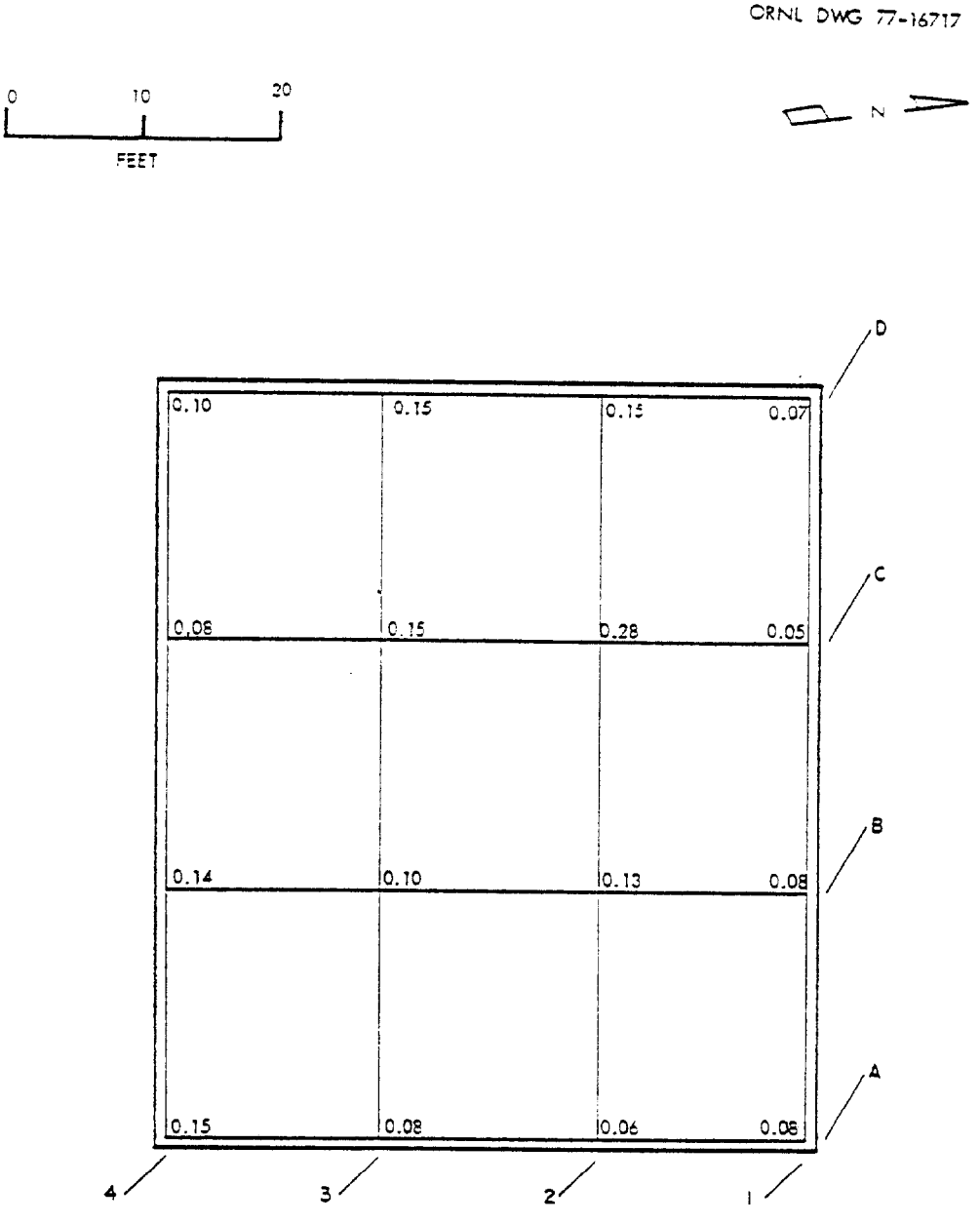

Fig. 7. Building 2, floor: beta-gamma dose rates (mrad/hr) at 1 cm above grid points.

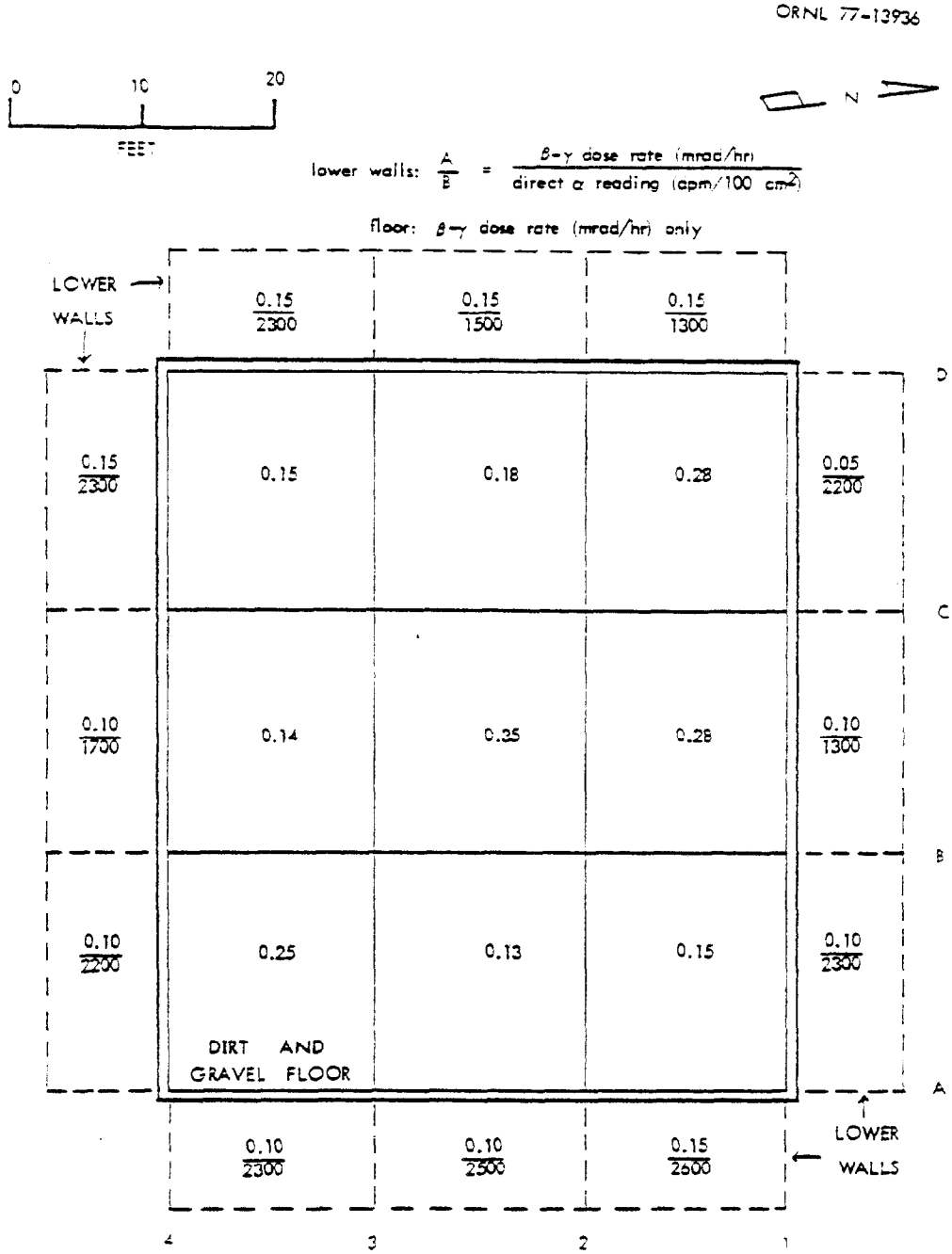

Fig. 8. Building 2, floor and lower walls: maximum observed beta-gamma dose rates and direct alpha readings.

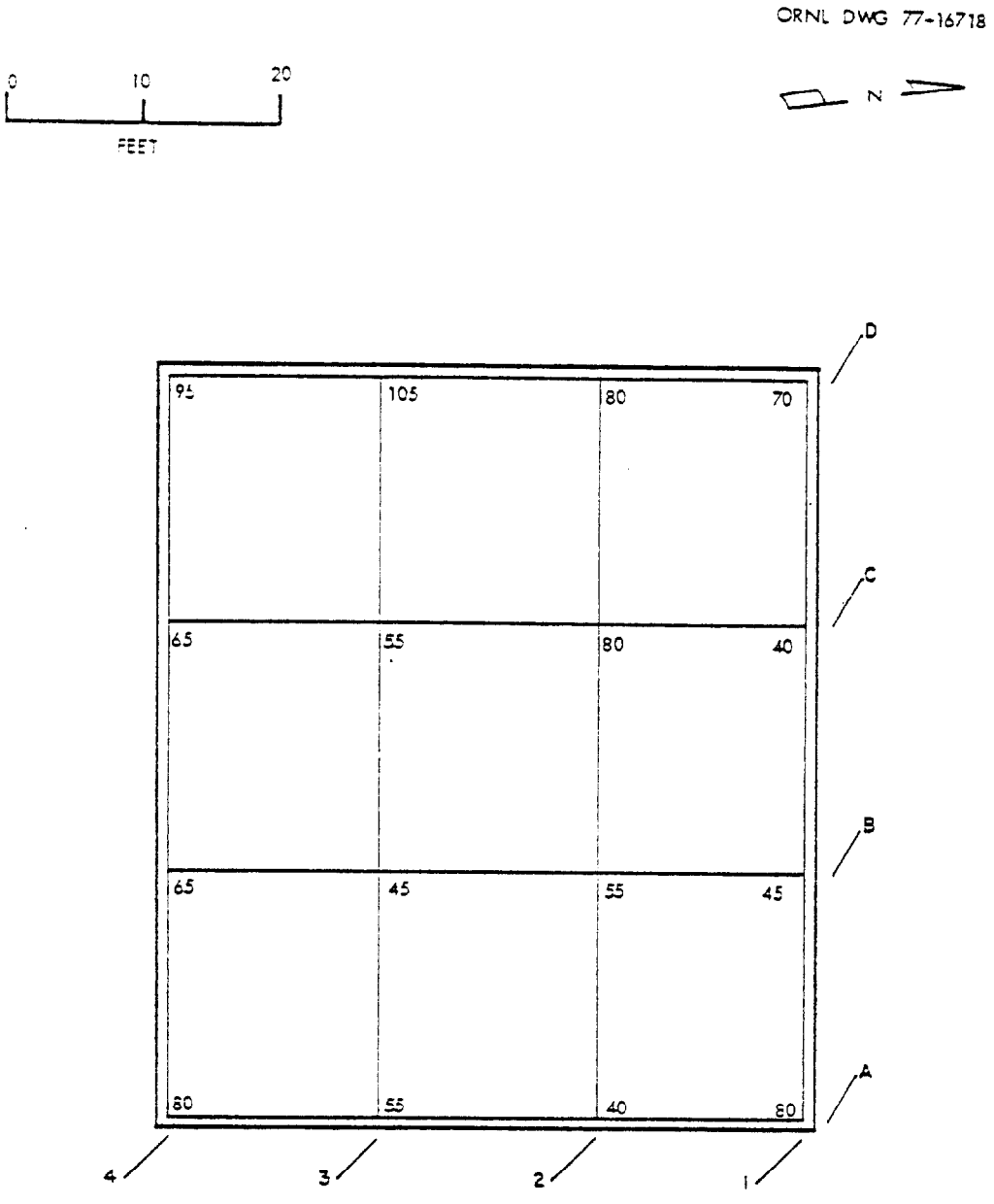

Fig. 9. Building 2: external gamma radiation levels (μR/hr) at 1 m above grid points.

ORNL 77-13939

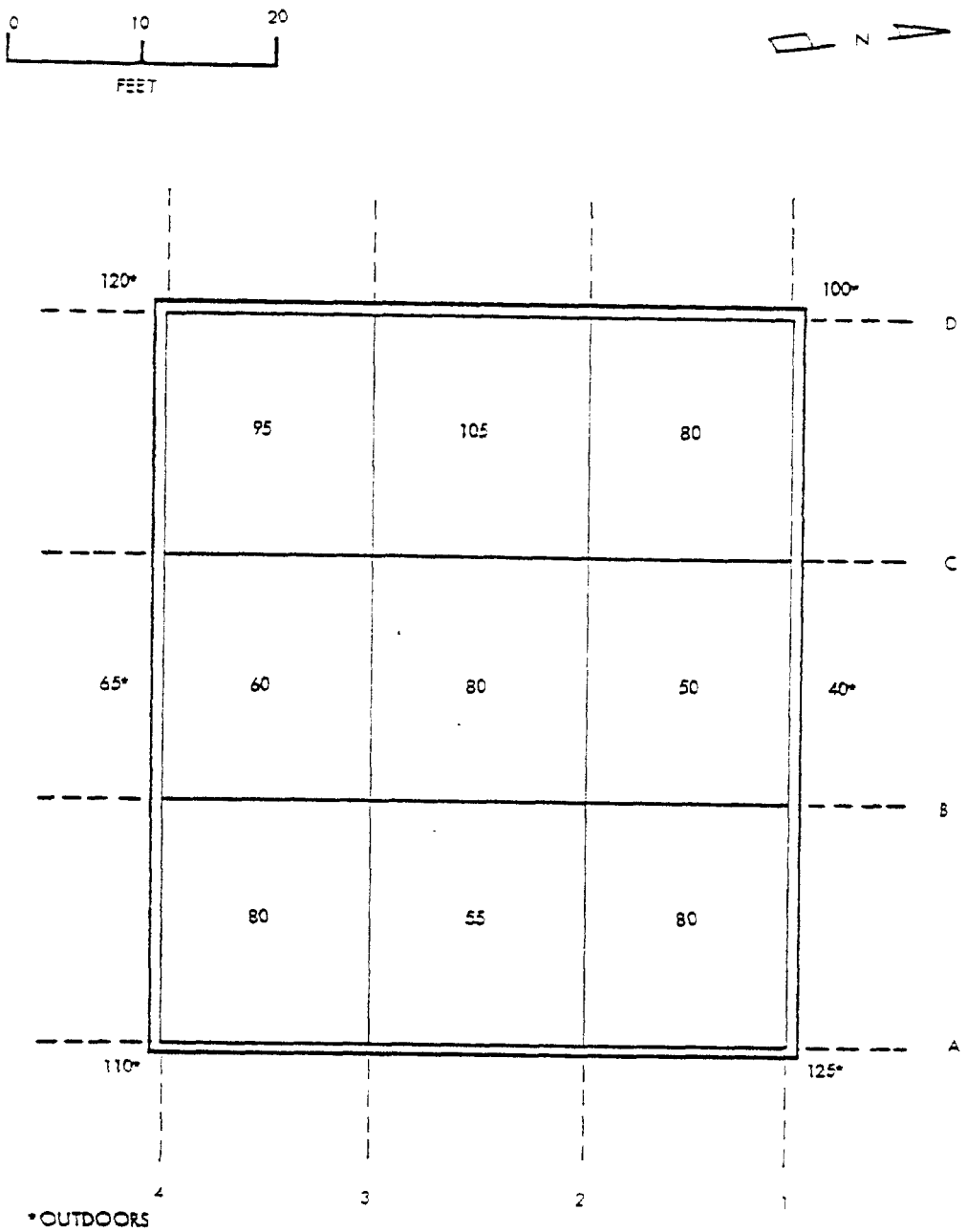

Fig. 10. Building 2, indoors and outdoors within 5 ft of walls: maximum external gamma radiation levels (in μR/hr) within survey blocks, at 1 m above floor.

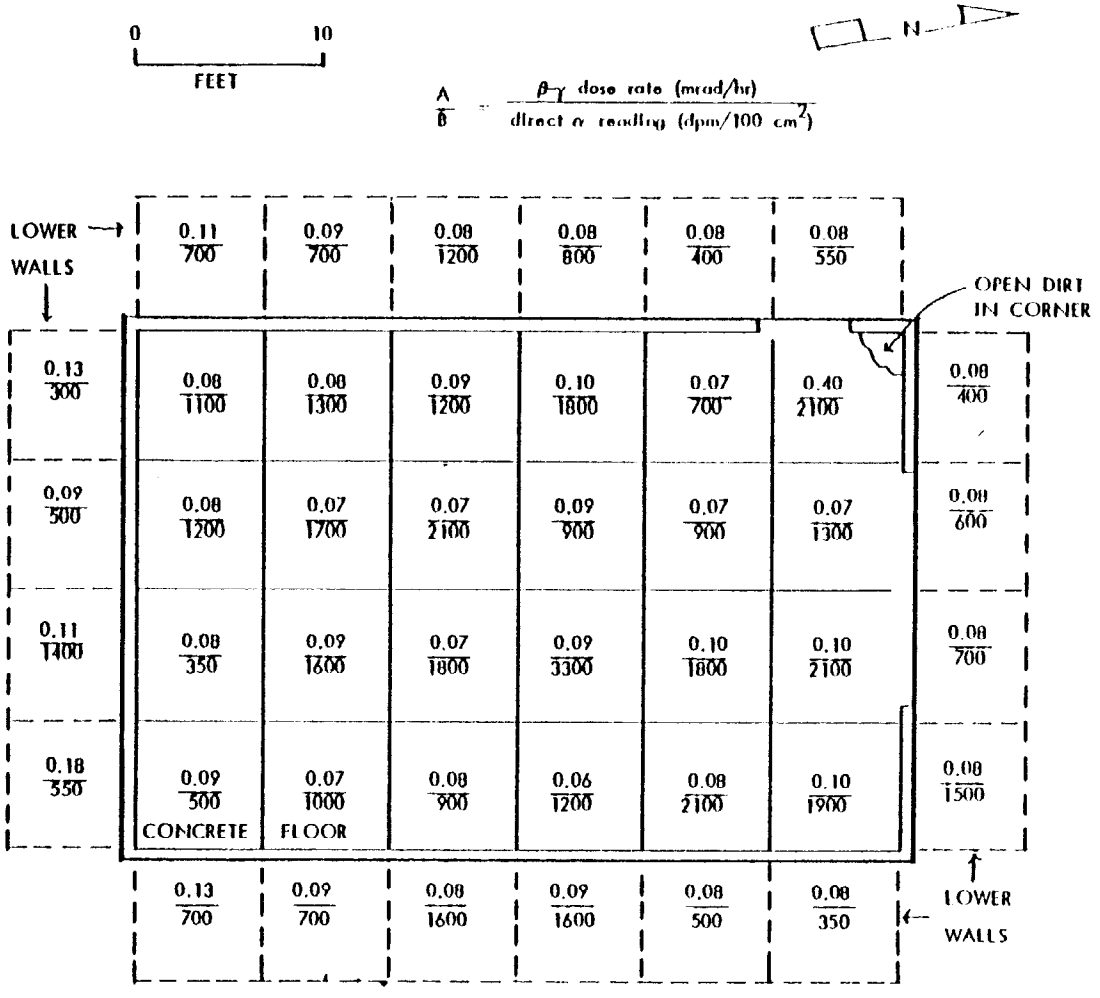

Fig. 11. Building 3, floor and lower walls: maximum observed beta-gamma dose rates and direct alpha measurements.

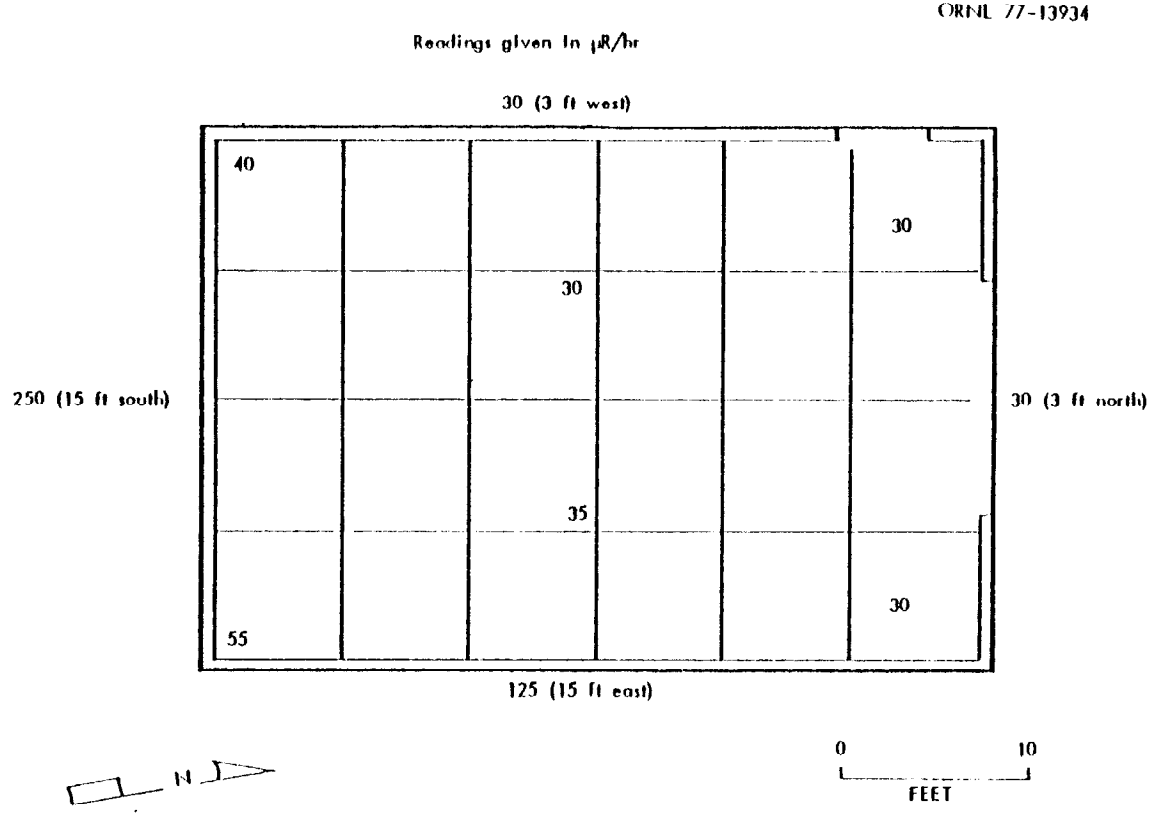

Fig. 12. Building 3, indoors and nearby outdoor points: external gamma levels (in μR/hr) at 1 m above surface.

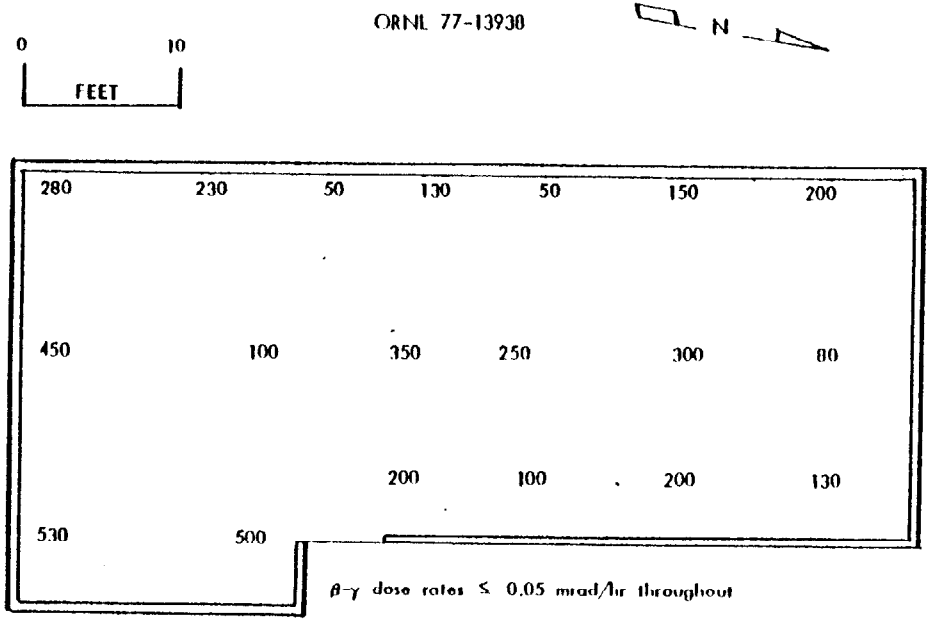

Fig. 13. Building 4, floor: direct alpha readings (in dpm/100 cm^2) at randomly selected points.

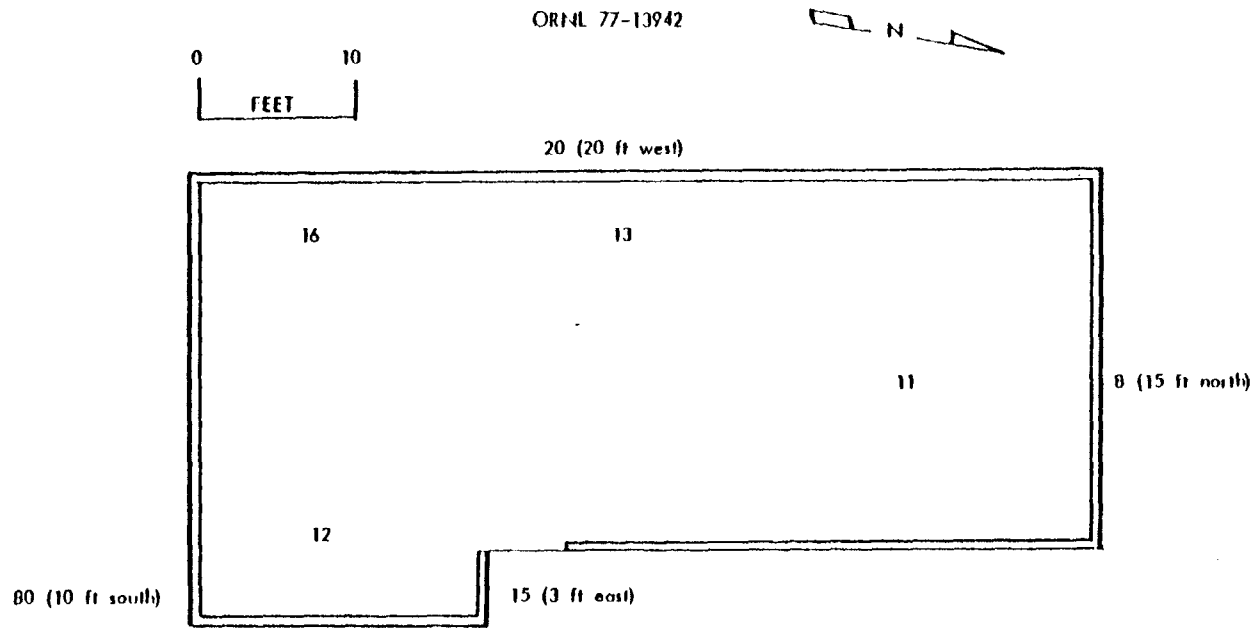

Fig. 14. Building 4, indoors and nearby outdoor points: external gamma radiation levels (in μR/hr) at 1 m above surface.

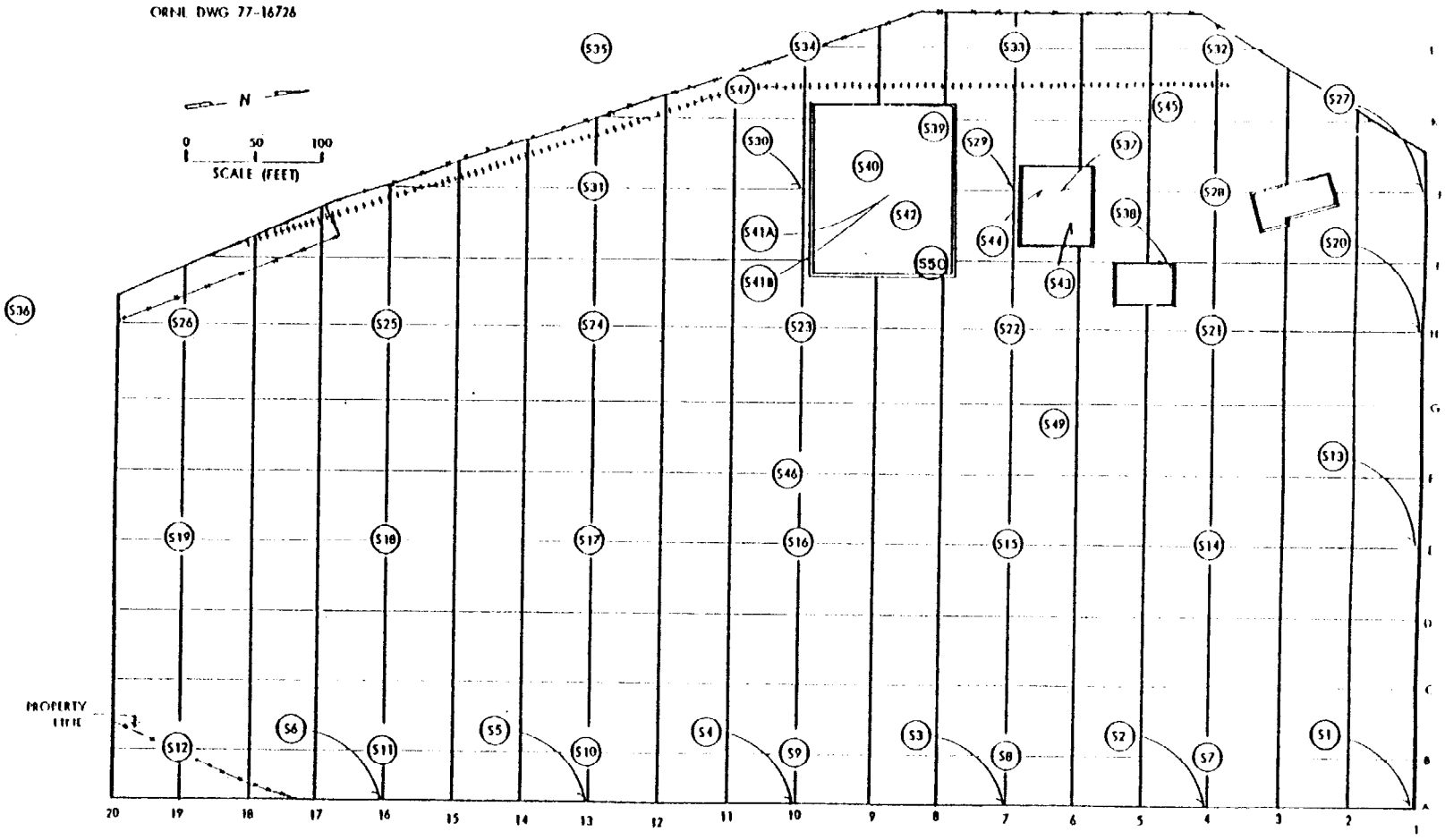

Fig. 15. Locations at which surface soil samples were taken.

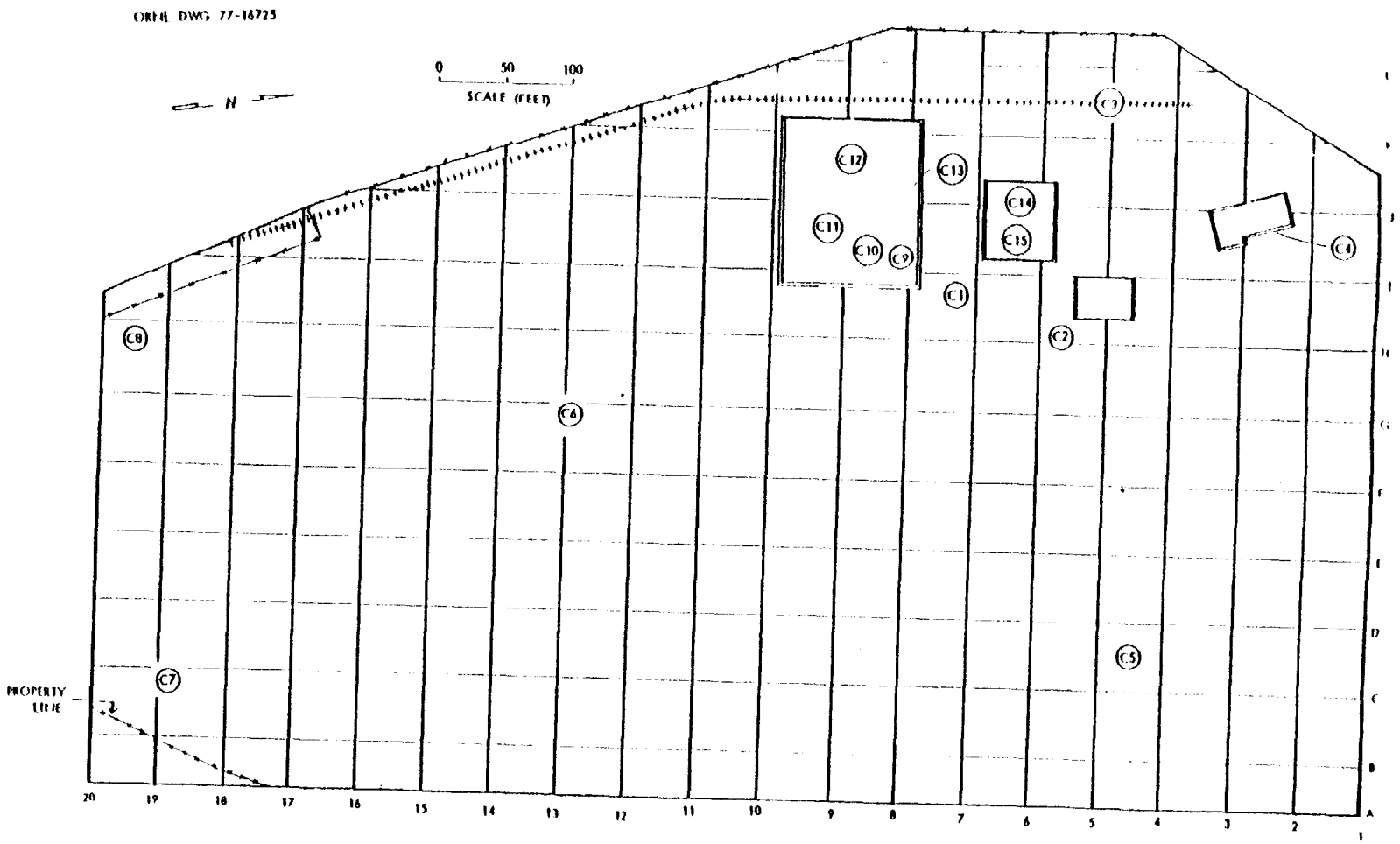

Fig. 16. Locations of core holes.

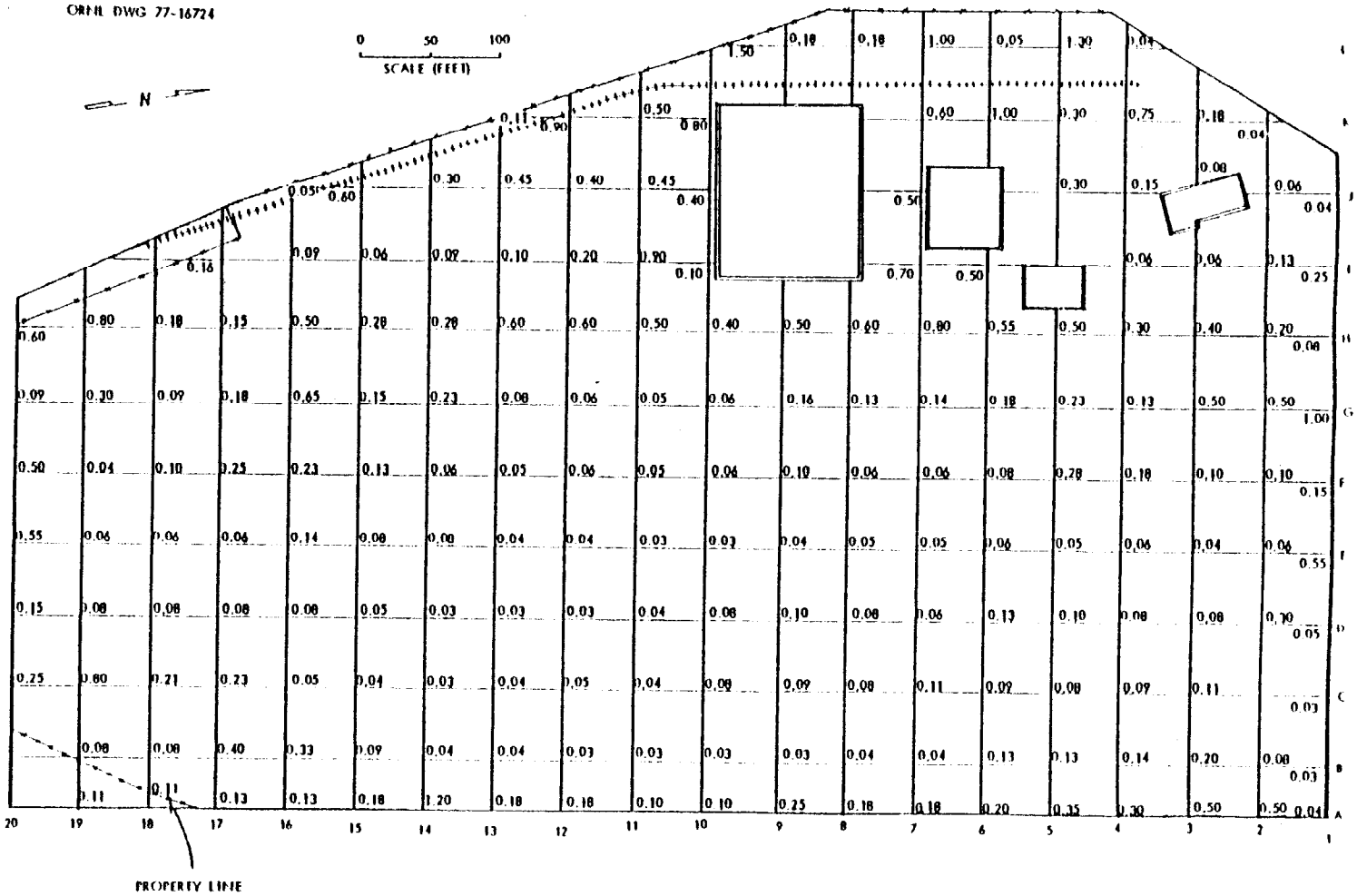

Fig. 17. Beta-gamma dose rates (in mrad/hr) outdoors at 1 cm above grid points.

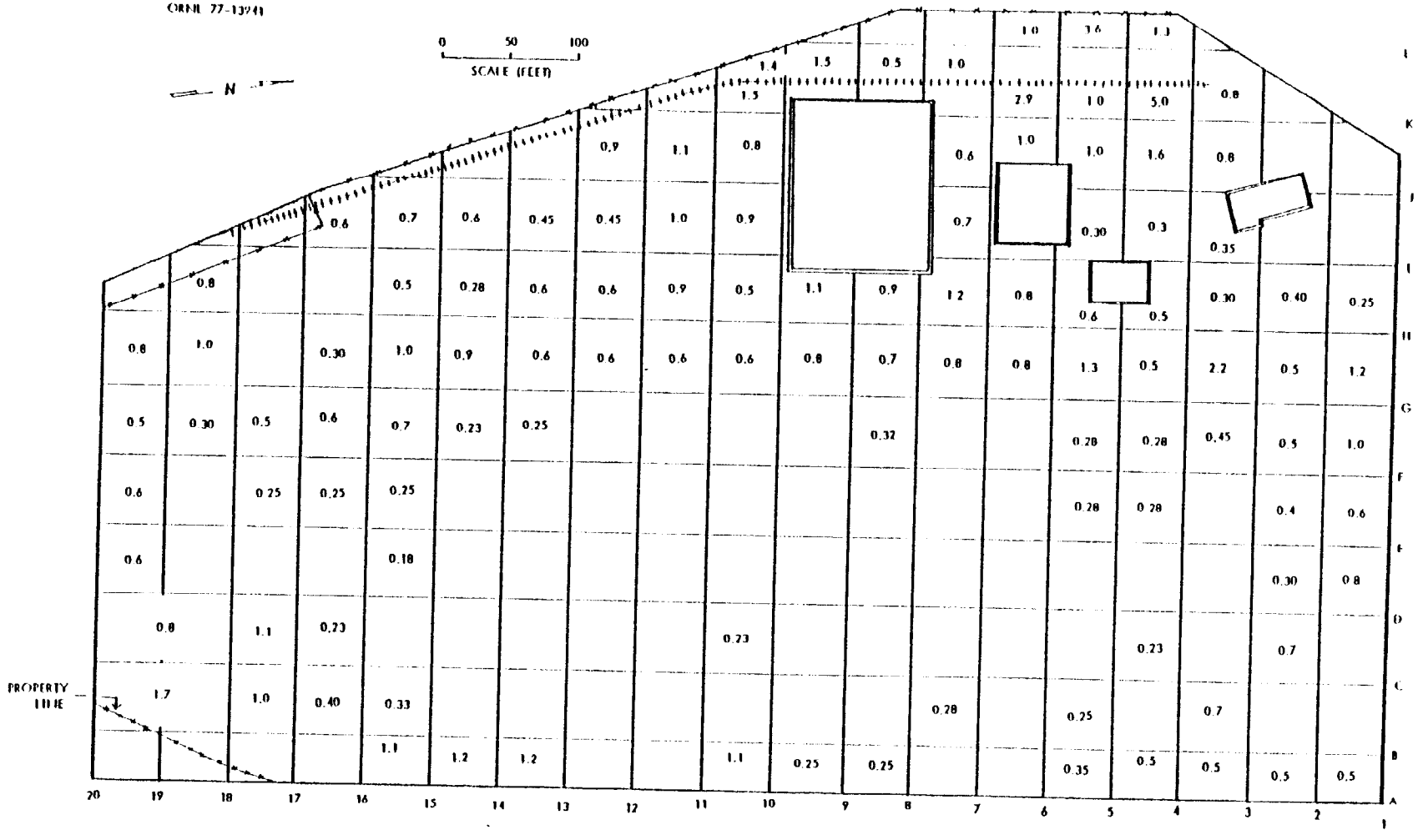

Fig. 18. Maximum observed beta-gamma dose rates (in mrad/hr) outdoors in those grid blocks where some readings exceeded 0.20 mrad/hr.

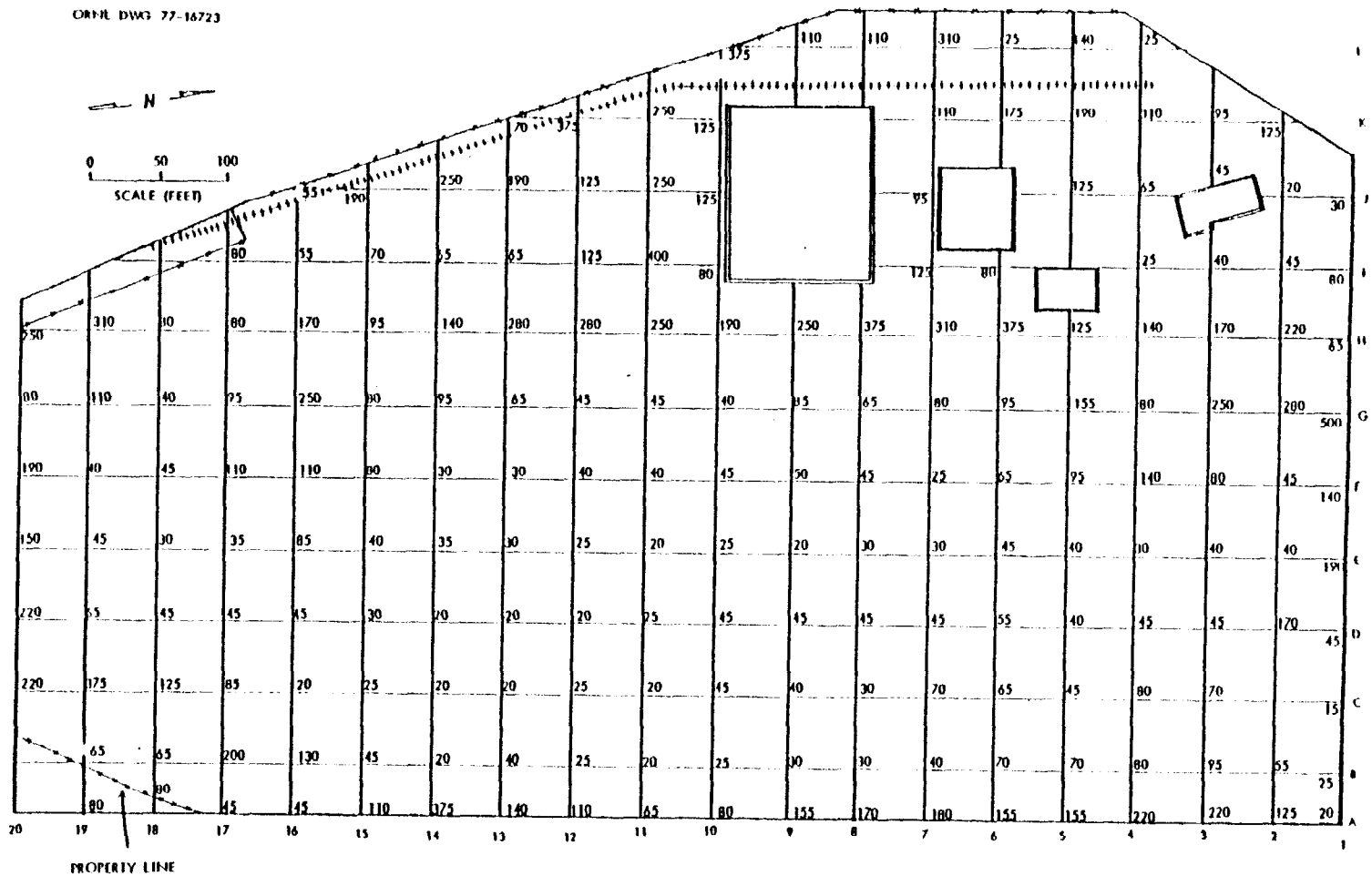

Fig. 19. External gamma radiation levels (in μR/hr) outdoors at 1 m above grid points.

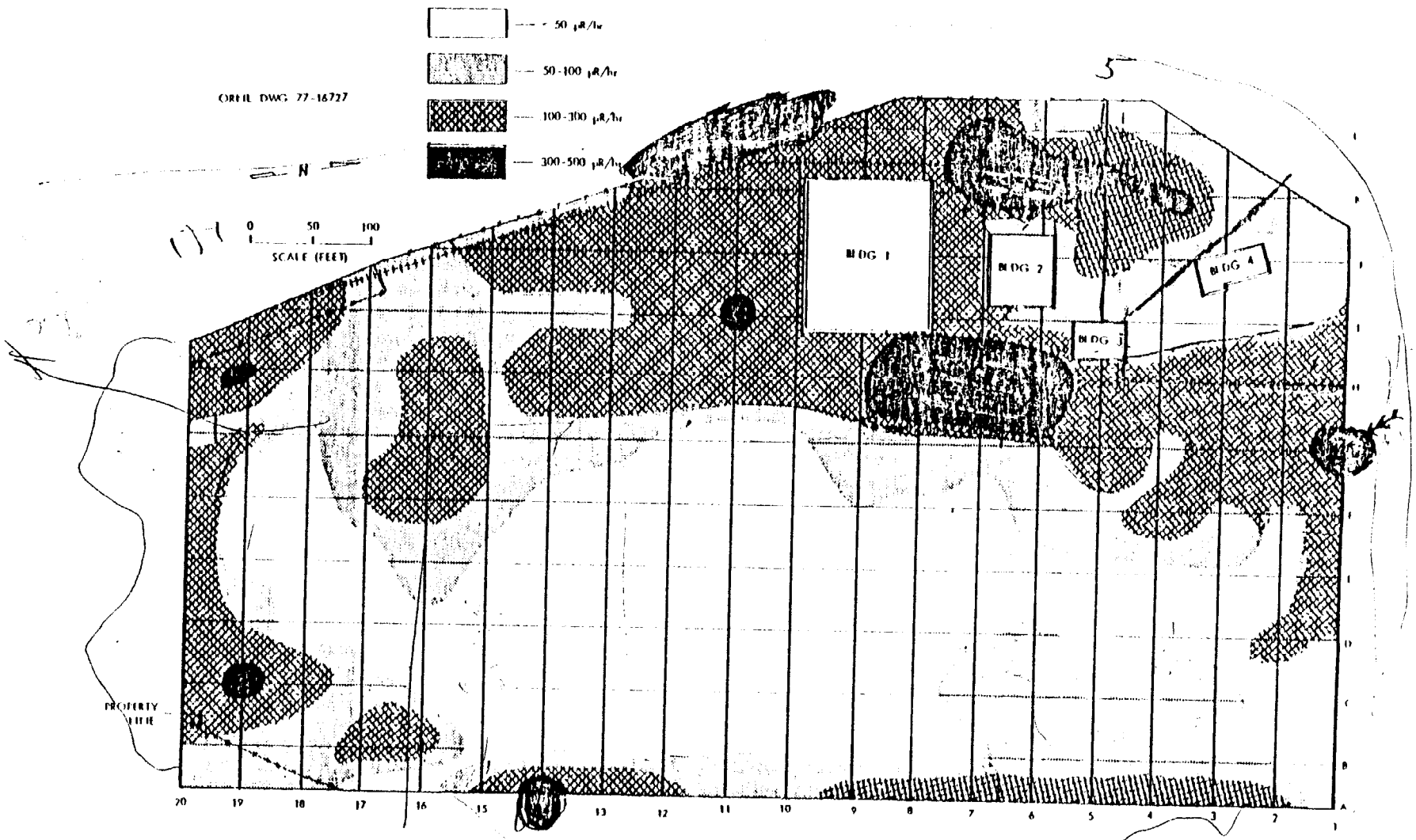

Fig. 20. Profile of average external gamma radiation levels (in μR/hr) at 1 m above surface outdoors.

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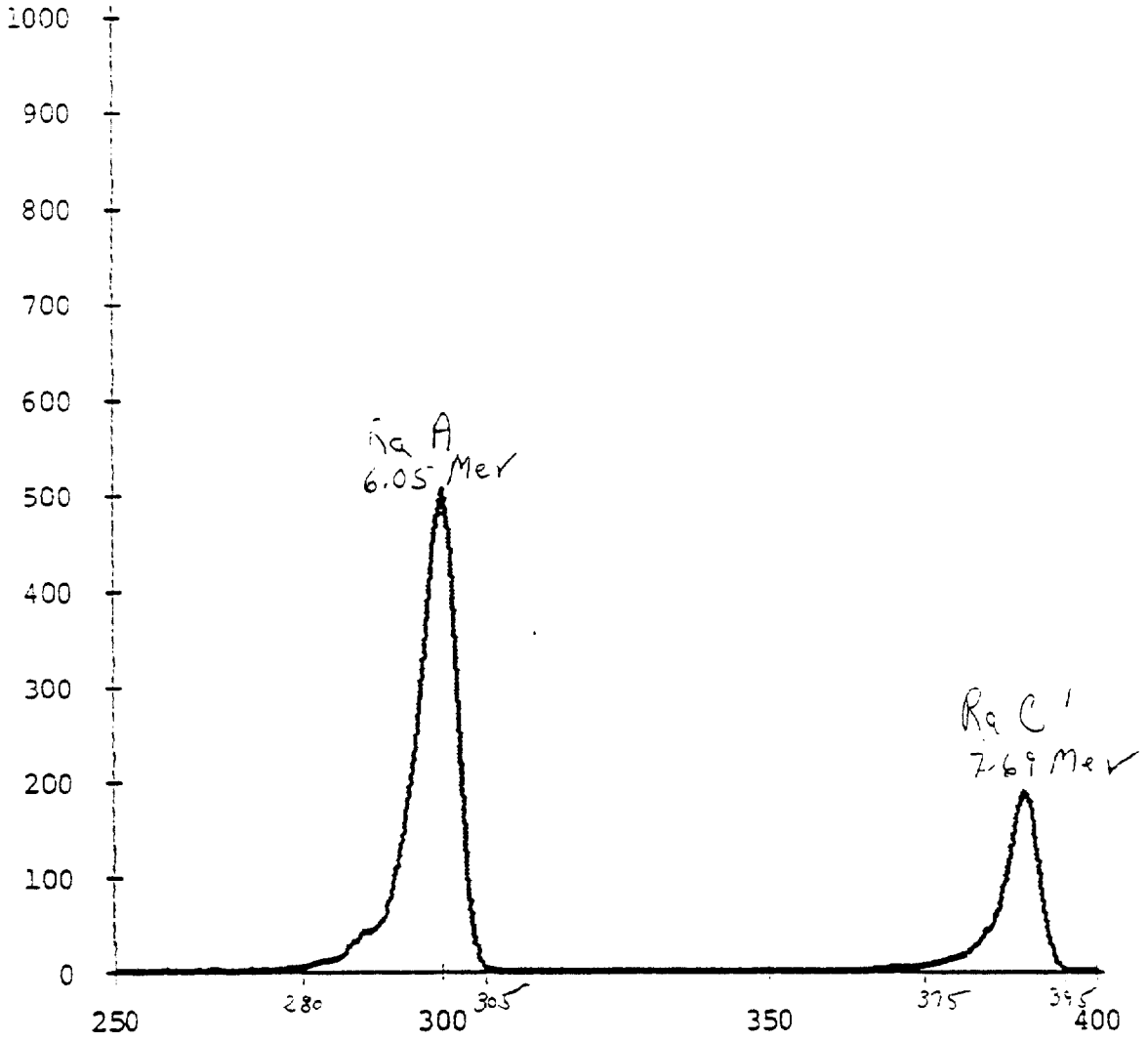

Fig. 21. A typical ^{222}Rn daughter alpha spectrum.

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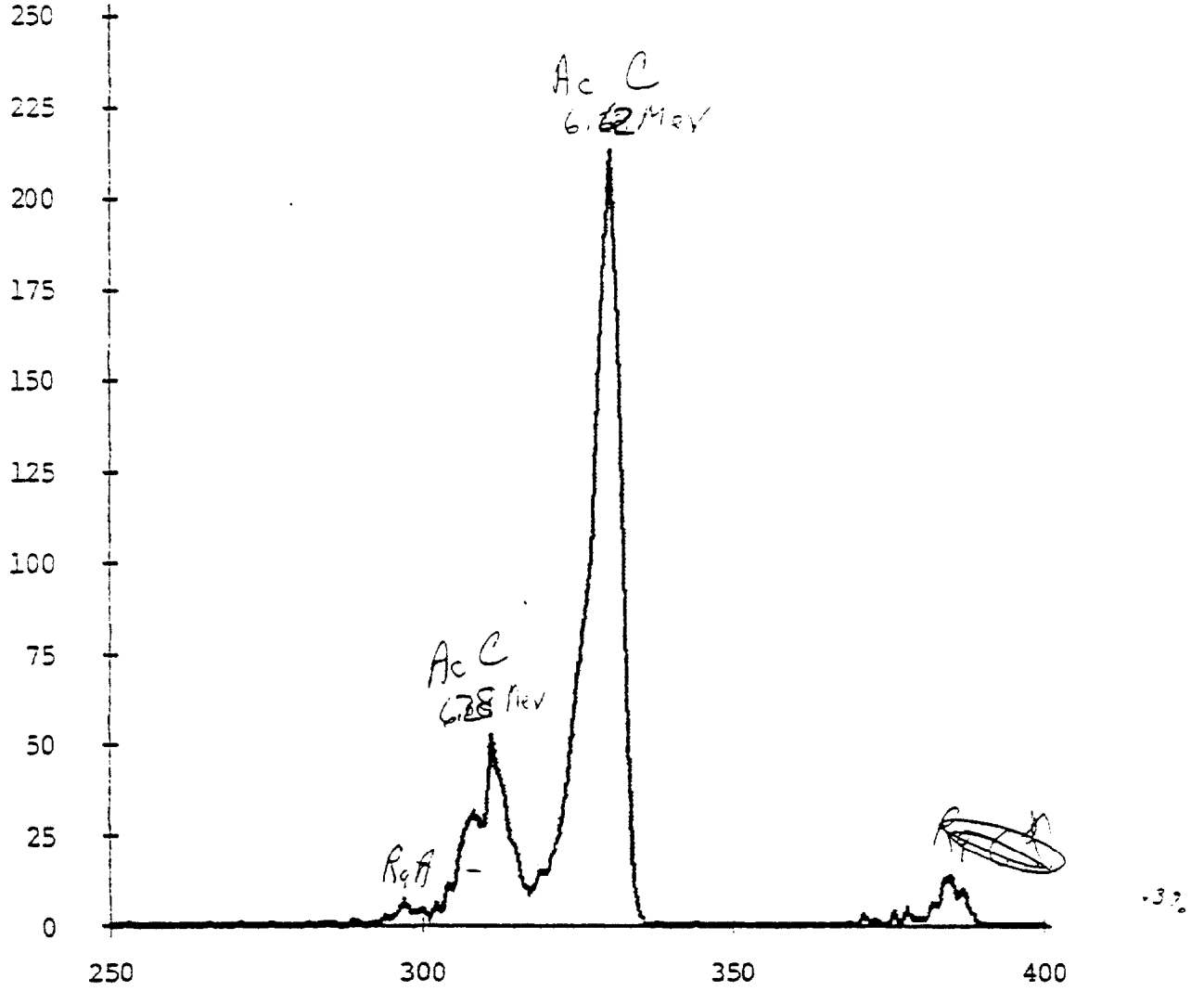

Fig. 22. Alpha spectrum for an air sample taken in Building 1.

Ac C ^{211}Bi 2.15 m 6.28 (16%)
6.62 (84%)

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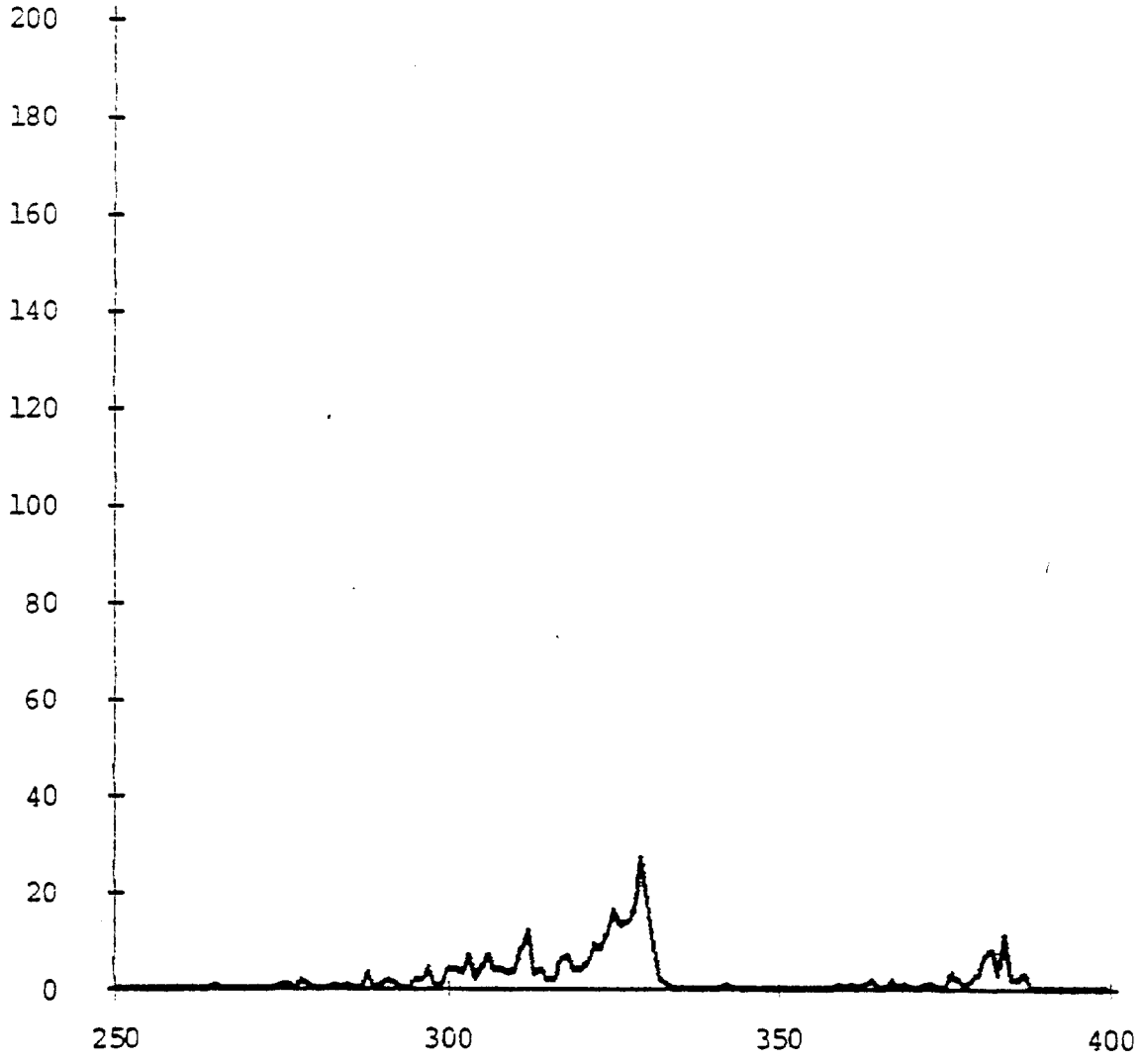

Fig. 23. Alpha spectrum for an air sample taken in Building 2.

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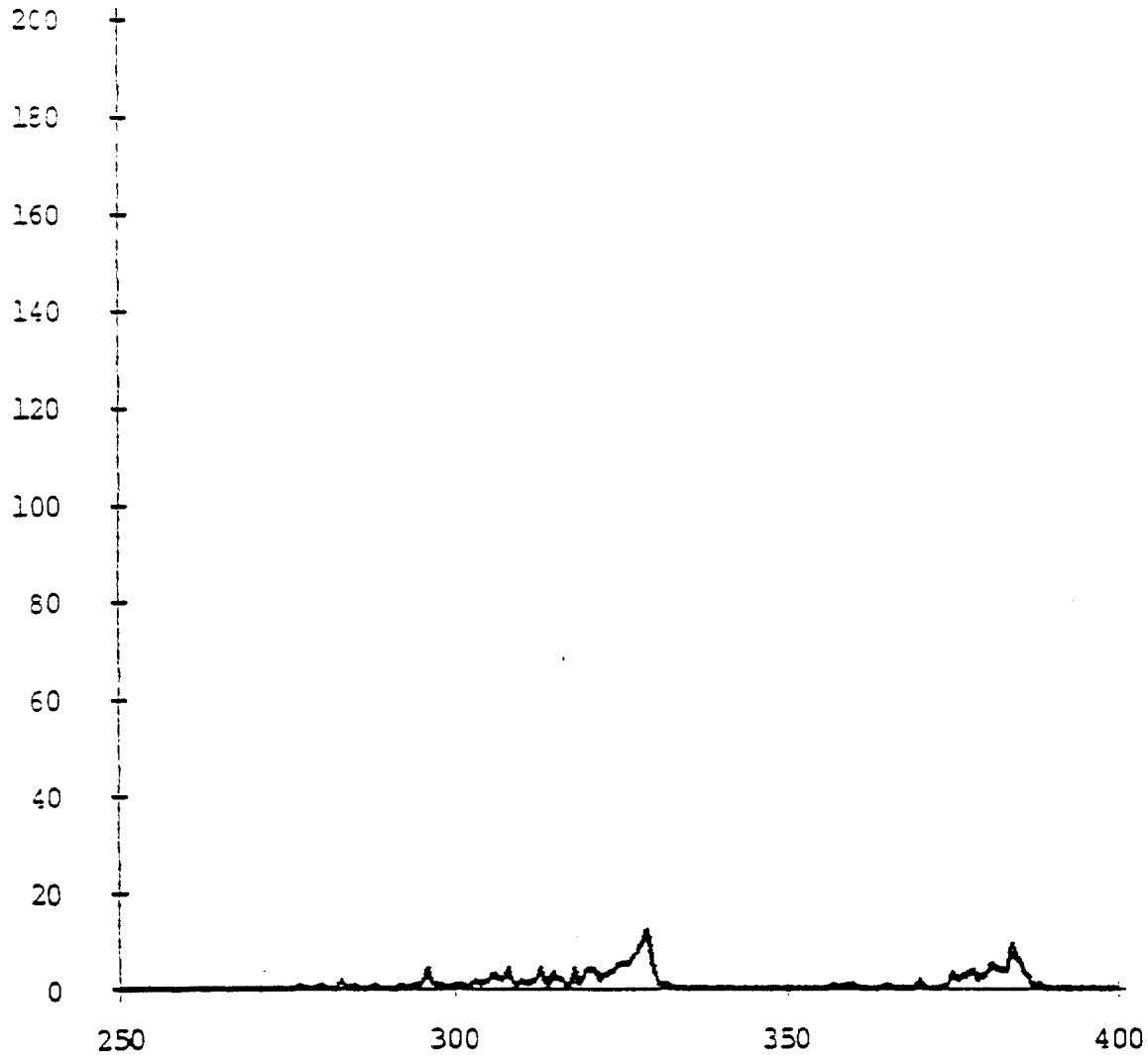

Fig. 24. Alpha spectrum for air sample taken in Building 3.

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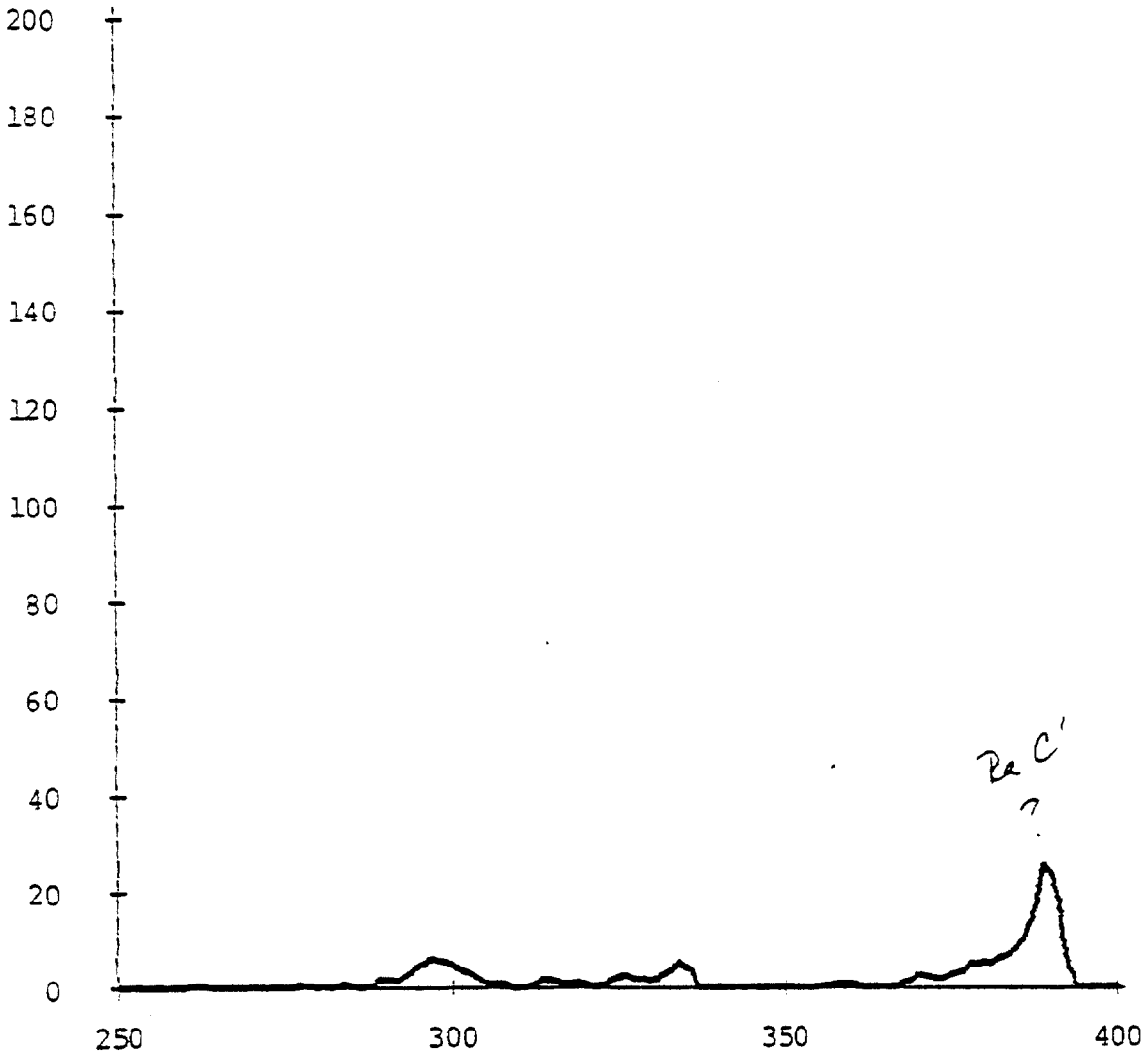


Fig. 25. Alpha spectrum for air sample taken in Building 4.

Table 1. Direct measurements of α and β-γ contamination levels on upper walls and ceiling in Buildings 1, 2, 3, and 4

Building	Number of measurements	Direct α measurements		β-γ dose rates	
		Average (dpm/100cm^2)	Maximum (dpm/100cm^2)	Average (mrad/hr)	Maximum (mrad/hr)
1	67	900	5500	0.24	0.60
2	36	280	1100	0.16	0.20
3	16	50	360	0.07	0.10
4	10	<50	100	0.03	0.05

Table 2. Transferable α and β contamination levels in Buildings 1, 2, 3, and 4

Building	Area	Number of smears	Transferable α contamination		Transferable β contamination	
			Average (dpm/100cm^2)	Maximum (dpm/100cm^2)	Average (dpm/100cm^2)	Maximum (dpm/100cm^2)
1	Walls, up to 6 feet from floor	39	115	450	45	250
1	Ceiling and walls above 6 feet	67	55	380	15	200
2	Walls, up to 6 feet from floor	26	50	140	20	110
2	Ceiling and walls above 6 feet	36	30	210	10	70
3	Floor	25	95	180	30	130
3	Walls, up to 6 feet from floor	40	30	100	20	100
4	Floor and walls	34	15	60	15	60

Table 3. Analyses of surface soil samples

Location[a]	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
S1	6.2	7.2	5.8	N.D.[b]
S2	230	310	320	N.D.
S3	35	52	N.D.	N.D.
S4	700	1000	640	N.D.
S5	14	21	7.1	2.1
S6	14	4.6	9.2	1.8
S7	69	100	78	N.D.
S8	88	25	37	N.D.
S9	3.3	5.5	1.0	1.3
S10	1.6	N.D.	N.D.	N.D.
S11	5.6	N.D.	N.D.	N.D.
S12	350	61	130	7.7
S13	830	810	1200	N.D.
S14	6.3	7.8	N.D.	N.D.
S15	N.D.	N.D.	N.D.	N.D.
S16	1.4	0.7	N.D.	1.2
S17	3.1	3.4	N.D.	N.D.
S18	89	N.D.	N.D.	N.D.
S19	14	7.4	7.0	1.3
S20	250	26	N.D.	N.D.
S21	20	N.D.	21	1.6
S22	160	N.D.	N.D.	N.D.
S23	160	240	93	N.D.
S24	220	426	180	N.D.
S25	310	N.D.	200	N.D.
S26	370	N.D.	N.D.	N.D.
S27	4.3	2.3	N.D.	N.D.
S28	130	N.D.	N.D.	N.D.
S29	54	647	35	N.D.
S30	130	N.D.	104	N.D.
S31	170	N.D.	N.D.	N.D.
S32	2.2	4.2	1.0	1.2
S33	8.9	N.D.	N.D.	N.D.
S34	250	330	N.D.	N.D.
S35	1300	N.D.	N.D.	N.D.
S36	2.4	N.D.	N.D.	N.D.
S37	80	84	47	N.D.
S38	82	3.1	120	N.D.
S39	430	860	530	N.D.
S40	320	550	370	N.D.

Table 3. (cont'd.) Analyses of surface soil samples

Location[a]	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
S41A	320	N.D.	370	5.2
S41B[c]	240	N.D.	240	8.6
S42	190	420	230	N.D.
S43	16	N.D.	10	0.44
S44	28	N.D.	16	0.6
S45	2700	N.D.	1500	N.D.
S46	3.0	N.D.	1.3	1.3
S47	470	530	390	N.D.
S48[d]	120	N.D.	110	4.5
S49	N.D.	210,000	N.D.	N.D.
S50	540	N.D.	700	N.D.

[a]Shown in Fig. 15.

[b]N.D. = concentration not determined.

[c]Sample taken at depth of 6 to 9 in.

[d]Sample taken from boots of surveyor who had walked in area shown in Fig. 2.

Table 4. Analyses of subsurface soil samples

Location[a]	Depth (in.)	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
C1	0-6	58	N.D.[b]	41	2.6
	6-12	95	N.D.	N.D.	N.D.
	12-18	130	N.D.	97	8.6
	18-24	2.1	N.D.	N.D.	1.2
C2	0-6	270	N.D.	140	6.3
	6-12	54	N.D.	22	1.1
	12-18	1.3	N.D.	N.D.	1.0
	18-24	2.2	N.D.	N.D.	N.D.
C3	0-6	1500	350	N.D.	N.D.
	6-12	25	N.D.	17	1.0
	12-18	12	N.D.	N.D.	N.D.
C4	0-6	8.8	N.D.	9.1	1.5
	6-12	1.7	N.D.	N.D.	N.D.
	12-18	1.9	N.D.	N.D.	N.D.
	18-24	N.D.	N.D.	N.D.	N.D.
C5	0-6	69	N.D.	34	3.1
	6-12	43	N.D.	N.D.	N.D.
	12-18	N.D.	N.D.	N.D.	N.D.
	18-24	49	150	37	3.5
C6	0-6	2.0	N.D.	N.D.	1.3
	6-12	12	N.D.	N.D.	N.D.
	12-18	1.9	N.D.	N.D.	N.D.
	18-24	1.0	N.D.	N.D.	1.3
C7	0-6	1100	N.D.	440	16
	6-12	820	N.D.	N.D.	N.D.
	12-18	11	N.D.	4.1	N.D.
	18-24	3.9	6.8	N.D.	1.3
C8	0-6	29	830	330	11
	6-12	44	N.D.	32	2.7
	12-18	2.4	9.8	1.0	1.3
	18-24	N.D.	N.D.	N.D.	N.D.
C9	0-6	50	70	N.D.	N.D.
C10	0-6	530	N.D.	540	24
	6-12	6.2	670	4.6	1.3
C11	0-6	30	N.D.	22	1.1
	6-12	1.5	1.8	N.D.	1.0
	12-18	1.5	1.7	N.D.	0.84
	18-24	1.4	1.8	0.91	1.1

Table 4. (cont'd.) Analyses of subsurface soil samples

Location[a]	Depth (in.)	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
C12	0-6	72	6.4	85	N.D.
	6-12	3.0	2.6	N.D.	0.95
	12-18	1.6	23	N.D.	1.3
	18-24	16	42	20	1.4
C13	0-6	140	220	65	4.3
	6-12	240	220	220	N.D.
	12-18	200	3.0	210	2.5
	18-24	3.0	190	N.D.	N.D.
C14	0-6	46	75	25	N.D.
	6-12	4.2	4.1	N.D.	1.2
	12-18	3.0	1.9	N.D.	N.D.
	18-24	1.9	2.0	N.D.	1.4
C15	0-6	34	38	240	2.0
	6-12	2.7	1.4	N.D.	1.1
	12-18	1.3	2.4	N.D.	1.2
	18-24	1.7	10	N.D.	N.D.

[a]Shown in Fig. 16.

[b]N.D. = concentration not determined.

Table 5. Radiochemical analyses[a] of selected samples

Sample location	^{230}Th (pCi/g)	^{228}Th (pCi/g)	^{232}Th (pCi/g)	^{231}Pa (pCi/g)	^{227}Ac (pCi/g)	^{226}Ra (pCi/g)
S45 Fig. 15	36,000	N.F.[b]	N.F.	900	900	present
S47 Fig. 15	18,000	N.F.	N.F.	900	<1800	present
From barrel found between grid points F19 and G19	90,000	9,000	9,000	<90	<90	9,000

[a] The activities listed here represent only the activities available by hot HCl leaching (normally between 50 and 100%).

[b] Not found.

Table 6. Concentrations of ^{210}Pb, ^{226}Ra, and ^{230}Th in water and sediment samples

Sample Location	Water			Sediment		
	^{210}Pb (pCi/ml)	^{226}Ra (pCi/ml)	^{230}Th (pCi/ml)	^{210}Pb (pCi/g)	^{226}Ra (pCi/g)	^{230}Th (pCi/g)
Outfall of property at Coldwater Creek	0.007 ± 0.001	0.002 ± 0.001	0.002 ± 0.001	55.9 ± 8.1	4.19 ± 1.35	91.9 ± 6.3
Coldwater Creek, ~2 miles downstream from site	<0.001	<0.001	0.0005 ± 0.0005	---------	0.063 ± 0.153	0.252 ± 0.248
Storm sewer at Latty Avenue	0.007 ± 0.003	0.001 ± 0.0005	<0.001	18.0 ± 5.0	<0.014	4.96 ± 0.90
RCG_W (soluble)	0.1	0.03	2.0	-	-	-

Table 7. Radon concentrations in air in Buildings 1, 2, 3, and 4

Building	Location[a]	Number of readings[b]	Maximum concentration (pCi/liter)	Average concentration (pCi/liter)
1	Near grid point D4	47	37.0	16.5
1	Near grid point B3	34	57.0	19.1
1	Near grid point F3	35	25.0	20.3
1	Near grid point E2	45	9.3	5.8
2	Between grid points B2 and B3	40	7.0	2.7
3	Center of building	40	1.0	0.5
4	Office, SE corner of building	88	5.8	1.9
4	Office, east side of building	88	3.7	1.5

[a]Refers to grid locations shown in Fig. 3 for Building 1, Fig. 7 for Building 2, and Fig. 11 for Building 3.

[b]Radon was measured continuously over periods of approxmately 24 hr; however, concentrations were recorded at intervals of 2000 sec. Because some radon and progeny from previous 2000 sec intervals remain in the Wrenn chamber, each reading actually represents a concentration which has been integrated over a period of 2 to 4 hr.

Table 8. Concentration guides listed in ERDA 0524 Annex A for uncontrolled areas

Nuclide	Compound	Air μCi/ml	Water μCi/ml
Ac 227	S[a]	8×10^{-14}	2×10^{-6}
Ac 227	I[b]	9×10^{-13}	3×10^{-4}
Pu 239	S	6×10^{-14}	5×10^{-6}
Pu 239	I	1×10^{-12}	3×10^{-5}
Ra 226	S	3×10^{-12}	3×10^{-8}
Ra 226	I	2×10^{-12}	3×10^{-5}
Pa 231	S	4×10^{-14}	9×10^{-7}
Pa 231	I	4×10^{-12}	2×10^{-5}
Ra 223	S	6×10^{-11}	7×10^{-7}
Ra 223	I	8×10^{-12}	4×10^{-6}
Th 230	S	8×10^{-14}	2×10^{-6}
Th 230	I	3×10^{-13}	3×10^{-5}

[a]Soluble.

[b]Insoluble.

APPENDIX I

DESCRIPTION OF RADIATION SURVEY

METERS AND SMEAR COUNTERS

RADIATION SURVEY METERS

Alpha Survey Meters

The type of alpha survey meter used at this site to measure alpha radioactivity on surfaces uses a ZnS scintillator to detect the alpha radiation.

The alpha scintillation survey meter consists of a large area (50 cm^2) ZnS detector with a photomultiplier tube in the probe which is coupled to a portable scaler/ratemeter (see Fig. I-A). The ZnS detector is covered with a 5-mil aluminized mylar sheet in order to make the instrument light-tight. A metal grid is used to avoid puncturing the mylar when surveying rough surfaces. This instrument is capable of measuring alpha surface contamination levels of a few dpm/100 cm^2 but must be used in the scaler mode for this purpose. It is highly selective for densely ionizing radiation such as alpha particles; the instrument is relatively insensitive to beta and gamma radiation.

Beta-Gamma Survey Meter

A portable Geiger-Muller (G-M) survey meter is the primary instrument for measuring beta-gamma radioactivity. The G-M tube is a halogen-quenched stainless steel tube having a 30 mg/cm^2 wall thickness and presenting a cross-sectional area of approximately 10 cm^2. Since the G-M tube is sensitive to both beta and gamma radiation, measurements are taken in both an open-window and a closed-window configuration. Beta radiation cannot penetrate the closed window, and, thus, the beta

reading can be determined by taking the difference between the open- and closed-window readings. This meter is shown in Fig. I-B.

The G-M survey meters were calibrated by comparison with a pre-calibrated Victoreen Model 440 ionization chamber (Fig. I-C). The open-window calibration factor was found to be 2000 cpm per mR/hr for surfaces contaminated with ^{226}Ra in equilibrium with ^{238}U and 2300 cpm per mR/hr for surfaces contaminated with initially pure uranium. The lower figure was routinely applied. The closed-window (gamma) calibration factor, determined by use of a NBS standard ^{226}Ra source, was 3200 cpm per mR/hr.

Gamma Scintillation Survey Meter

A portable survey meter using a NaI scintillation probe is used to measure low-level gamma radiation exposure. The scintillation probe is a 3.2 x 3.8-cm NaI crystal coupled to a photomultiplier tube. This probe is connected to a Victoreen Model Thyac III ratemeter (see Fig. I-D). This unit is capable of measuring radiation levels from a few μR/hr to several hundred μR/hr. This instrument is calibrated at ORNL with an NBS standard ^{226}Ra source. Typical calibration factors are of the order of 300 cpm/μR per hr.

SMEAR COUNTERS

Alpha Smear Counter

This detector assembly, used for the assay of alpha emitters on smear paper samples, consists of a light-tight sample holder, a zinc sulfide phosphor and a photomultiplier tube. This detector assembly was used with electronic components housed in a portable NIM bin (see Fig.

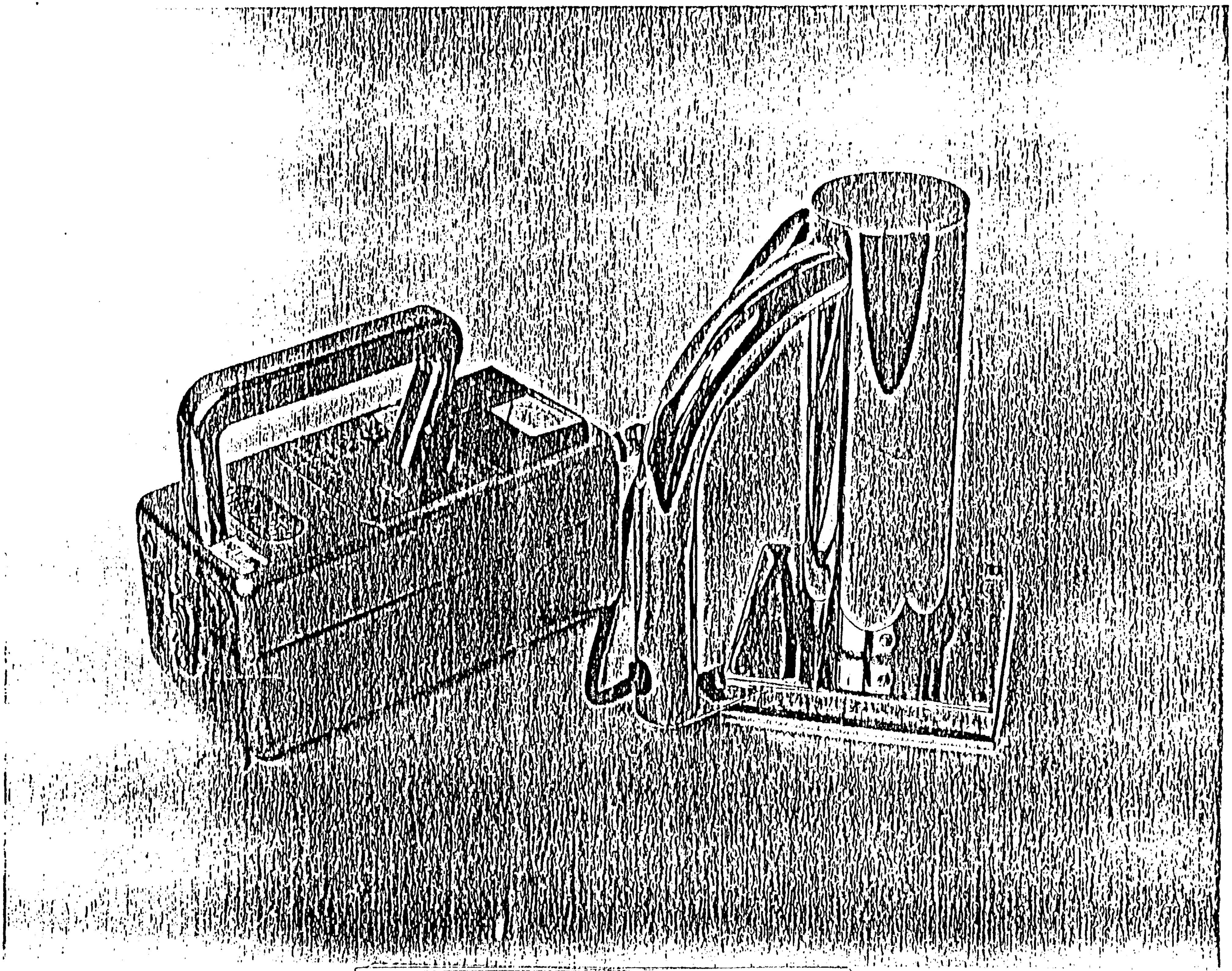
I-E). The electronics package consisted of a preamplifier, a ORTEC 456 high voltage power supply, a Tennelec TC 211 linear amplifier and a Tennelec TC 545 counter-timer.

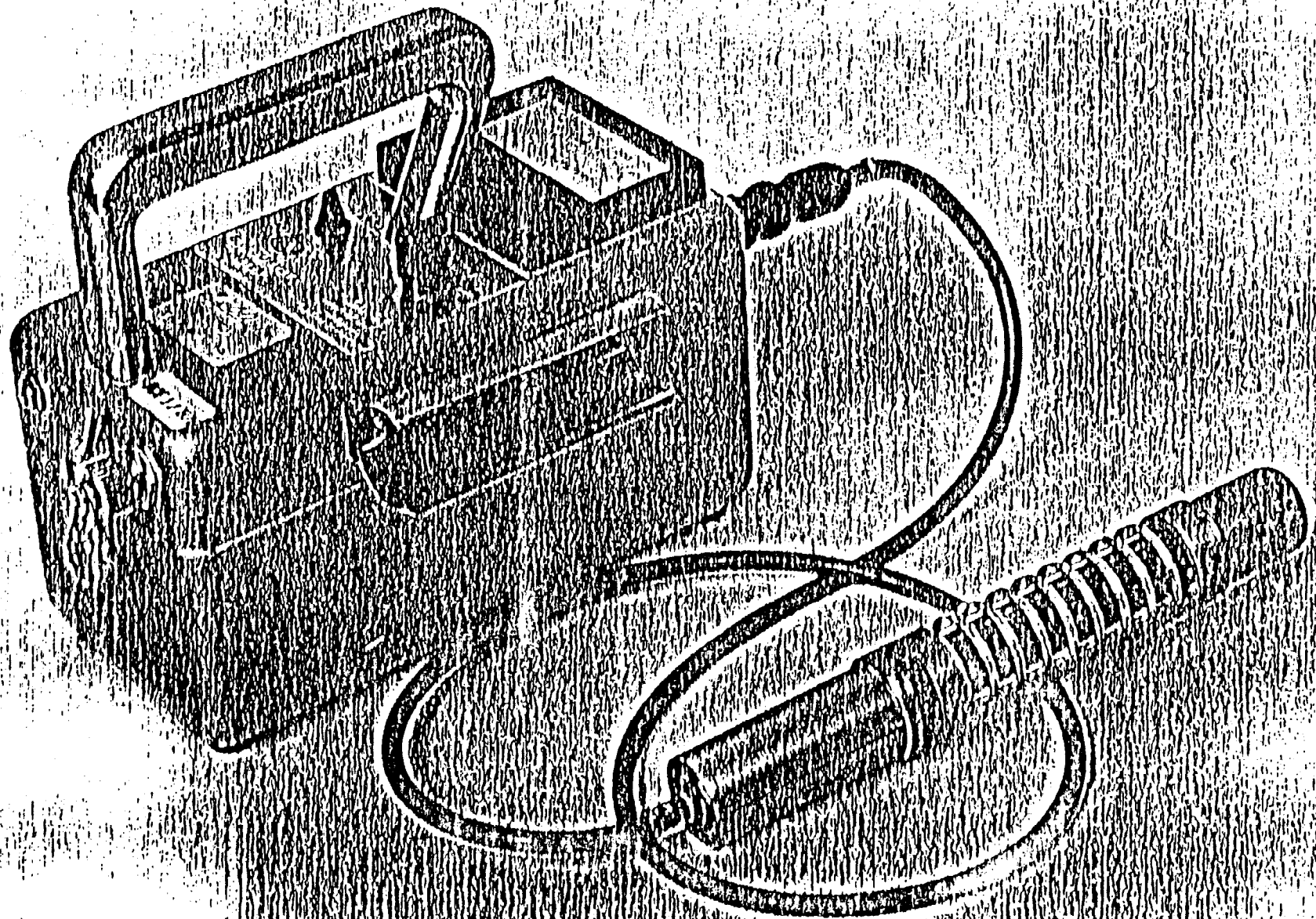
The alpha smear counter was used in the field and was calibrated daily using an alpha source with a known disintegration rate.

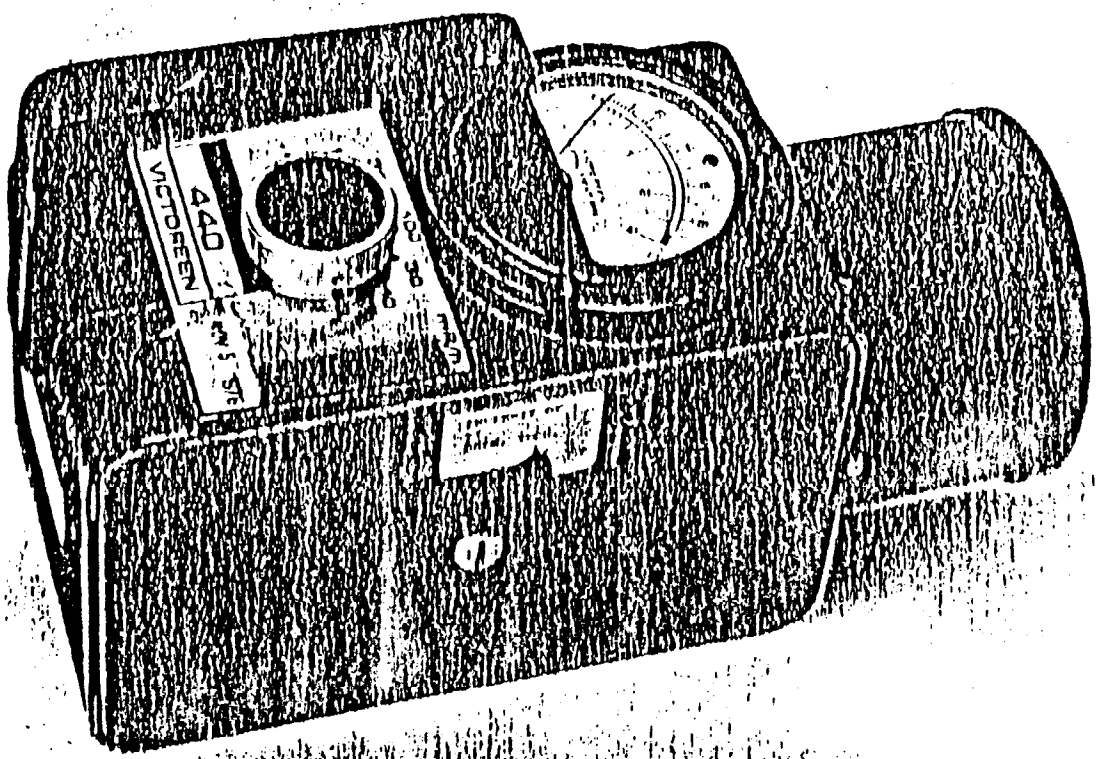
Beta Smear Counter

The beta smear counter consisted of a thin mica window ($\sim$ 2 mg/cm^2) G-M tube mounted on a sample holder and housed in a 23-cm diam x 35-cm high lead shield. Located under the counter window is a slotted sample holder, accessible through a hinged door on the shield. An absorber can be interposed in the slot between the sample and the counter window to determine relative beta and gamma contributions to the observed sample counting rate. The electronics for this counter were housed in a portable NIM bin and consisted of a Tennelec TC 148 preamplifier, an ORTEC 456 high voltage power supply and a Tennelec TC 545 counter-timer.

This unit, shown in Fig. I-E, was used in the field to measure beta activity on smear papers and was calibrated daily using a beta standard of known activity. Since the beta smear counter usually shows a relatively high background (typically 12 to 20 cpm) and has a relatively low efficiency ($\sim$ 6%), accurate measurement of very low-level beta contamination on a single smear would require a count of several minutes. Because of the large number of smear samples to be counted on each site, each sample is counted for one minute, and accuracy is improved for low-level counts by averaging results for several smears. For example, if 25 smear samples show an average transferable beta contamination level of 30 dpm/100 cm^2, there is a probability of approximately 68% that the actual average contamination level is at least 20 dpm/100 cm^2.







Victoreen Model 440 Ionization

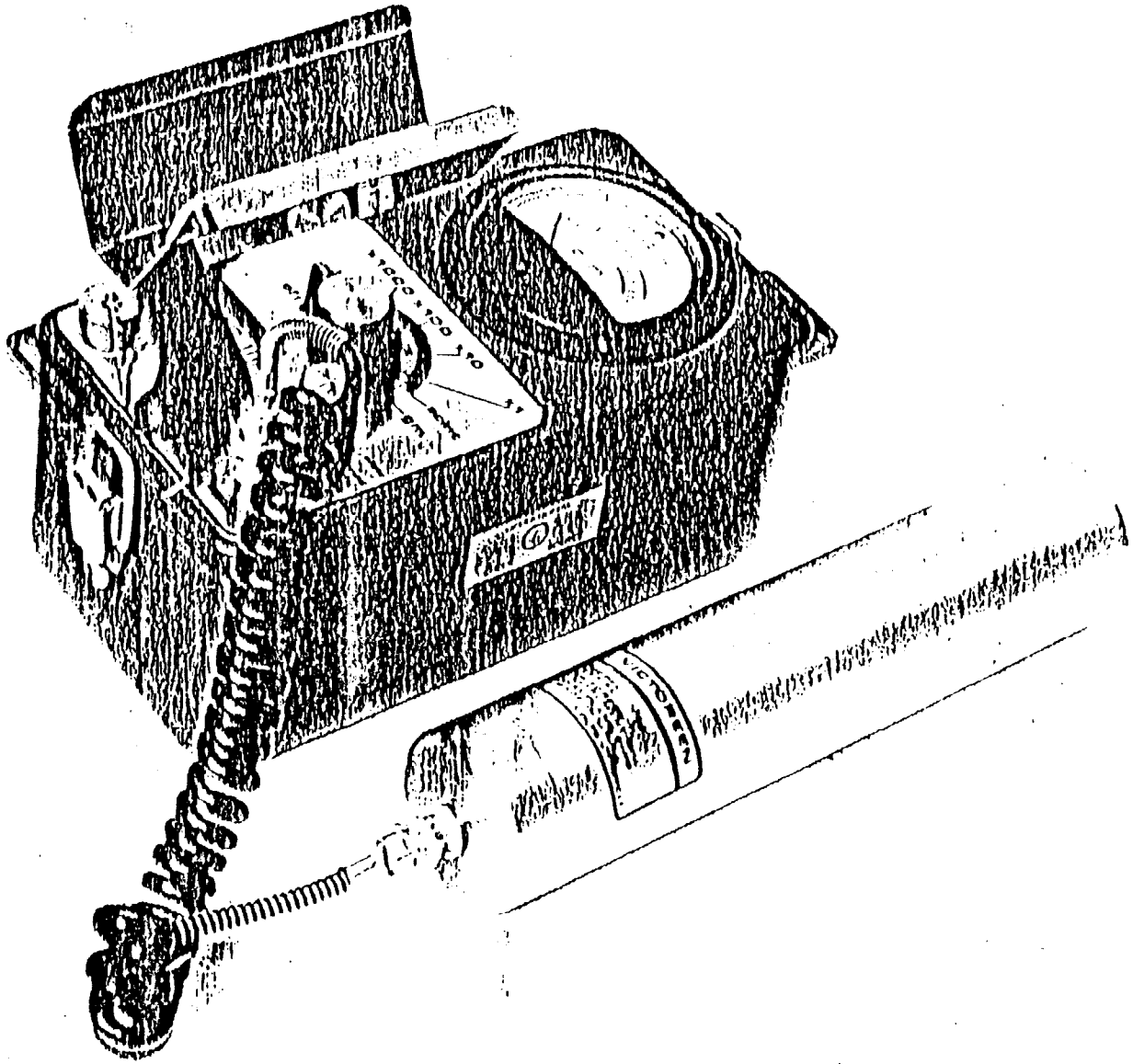


Fig. 1-2. Gamma scintillation survey meter.

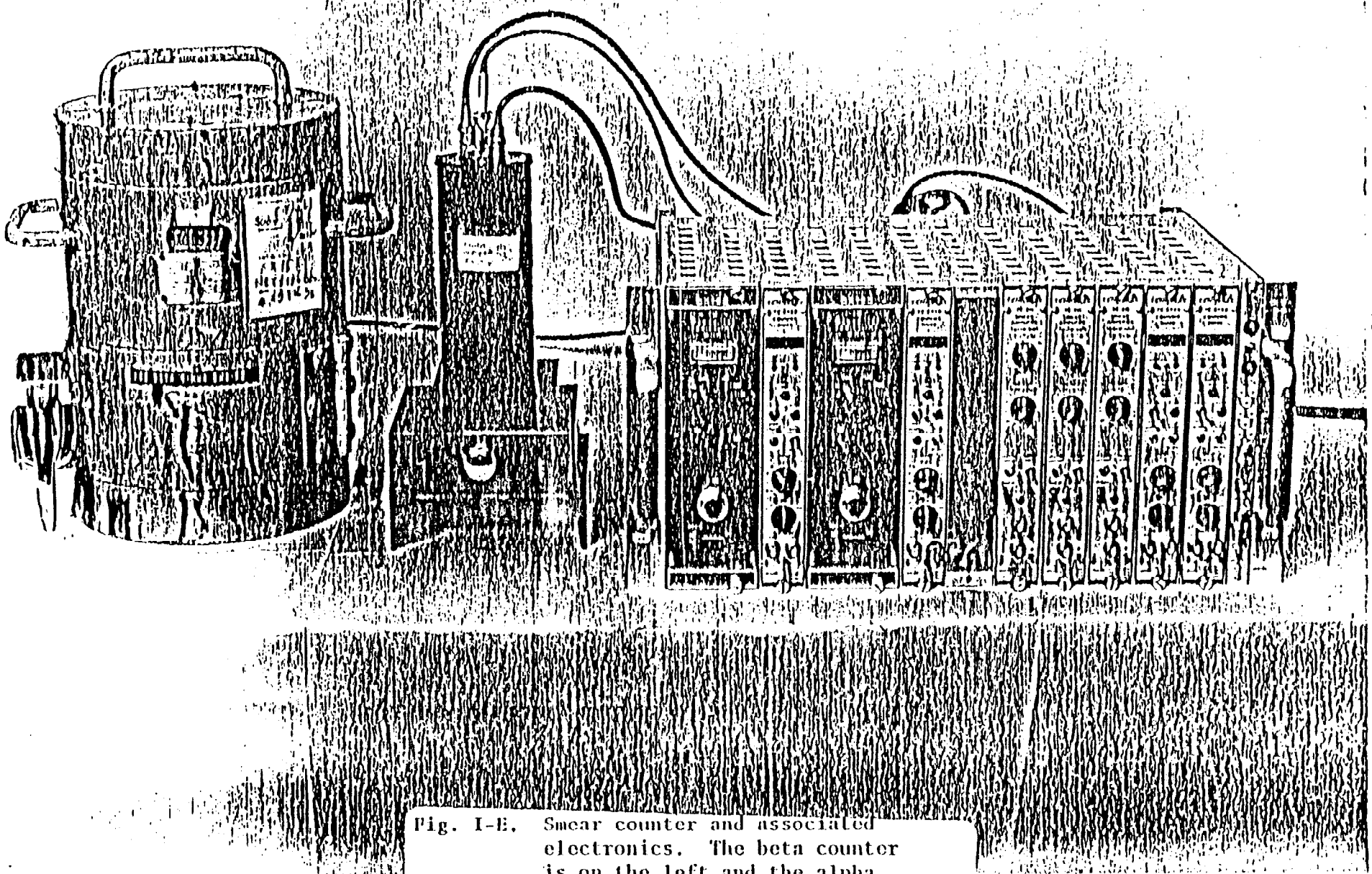


Fig. I-E. Smear counter and associated electronics. The beta counter is on the left and the alpha

APPENDIX II

TECHNIQUES FOR THE MEASUREMENT OF RADON AND RADON DAUGHTER CONCENTRATIONS IN AIR

TECHNIQUE FOR THE MEASUREMENT OF ^{222}Rn PROGENY CONCENTRATIONS IN AIR

An alpha spectrometry technique has been developed for the measurement of ^{222}Rn progeny concentrations in air. From one integral count of the ^{218}Po alpha activity and two integral counts of the ^{214}Po alpha activity, the concentration in air of ^{218}Po, ^{214}Bi, and ^{214}Pb may be calculated.

Particulate ^{222}Rn daughters attached to airborne dust are collected on a membrane filter with a pore size of 0.4 microns. A sampling time of 5 min and a flow rate of 12 LPM are used. This filter sample is then placed under a silicon surface barrier detector and counted. The detector and counting system used for radon daughter measurements are shown in Fig. II-A. Usually, counting of this kind is performed with a vacuum between the sample and the detector which requires a complicated sample holder and time-consuming sample changing methods. Experiments at this laboratory have shown that ease in sample handling is obtained with little loss in resolution when helium is used as a chamber fill gas. In this counter, helium is flowed between the diode and the filter sample, which are separated by a distance of 0.5 cm. One integral count of the ^{218}Po alpha activity is obtained from 2 to 12 min, and two integral counts of the ^{214}Po activity are obtained from 2 to 12 min and 15 to 30 min, respectively. All counting intervals are referenced to t = 0 at the end of sampling.

The equations describing the ^{222}Rn progeny atoms collection rates on the filter are of the form

$$\frac{dn_i(t)}{dt} = C_i v + \lambda_{i-1} n_{i-1}(t) - \lambda_i n_i(t) \qquad (1)$$

where

n_i = number of the i^{th} species of atom on the filter as a function of time,

λ_i = radioactive decay constant of the i^{th} species (min^{-1}),

C_i = concentration of the i^{th} species (atoms l^{-1}), and

v = air sampling flow rate (liters min^{-1}).

The solution of Eq. (1) is of the form

$$y = e^{-ax} \left[y_0 = \int F(x)\, e^{ax} dx\right].$$

From the general form of the solution, specific equations can be obtained describing the number of each ^{222}Rn decay product collected on the filter as a function of time. Also by letting $v = 0$ in Eq. (1), a set of equations describing the decay on the filter of each ^{222}Rn progeny can be obtained. The equations describing the decay of ^{222}Rn progeny on the filter can be integrated and related to the integral counts obtained experimentally. Values for the total activities of ^{218}Po, ^{214}Pb, and ^{214}Bi on the filter at the end of sampling are obtained by applying matrix techniques. The airborne concentrations are obtained by solving the equations describing the atom collection rates on the filter. A computer program has been written to perform these matrix operations, to calculate the air concentrations of the radon progeny, and to estimate the accuracy of the calculated concentrations.

The procedure described above was applicable in Building 4 on the Latty Avenue site but was not applicable in Buildings 1, 2, and 5 because of the presence of airborne ^{220}Rn and ^{219}Rn in those structures.

The alpha energies of some daughters of these radionuclides are nearly indistinguishable (by alpha spectroscopy methods) from the alpha energies of some daughters of ^{222}Rn. However, ^{211}Bi, a daughter of ^{219}Rn, was distinguished from daughters of ^{222}Rn and ^{220}Rn by taking two 10-min counts separated by an interval of 50 min. Also, the concentration of Th-B and Th-C were estimated by counting immediately after sampling and again at 5 hr after sampling.

TECHNIQUE FOR THE MEASUREMENT OF RADON CONCENTRATIONS IN THE AIR

Wrenn Chambers were used for the measurement of ^{222}Rn concentrations in air. The Wrenn Chamber operates on the principle that RaA ions are positively charged. Radon is allowed to diffuse through a foam-rubber-covered, hemispherically shaped metal screen, which filters radon daughters. As radon decays, after diffusing into the cavity, RaA ions are attracted to a thin aluminized mylar film which is stretched over a zinc sulfide scintillation detector. The potential between this aluminized mylar film and the hemispherically shaped wire screen creates a strong electric field which serves to attract the charged ions. The ions thus attracted remain on the surface of the mylar film and continue their radioactive decay to other radon daughters. The principal radiation detected by a radon monitor of this type is the alpha particles from RaA and RaC'. The Wrenn Chambers are calibrated by using a known radon source.

APPENDIX III

DESCRIPTION OF Ge(Li) DETECTOR AND SOIL COUNTING PROCEDURES

DESCRIPTION OF Ge(Li) DETECTOR SYSTEM

A holder for twelve 30-cm^3 polyethylene bottles (standard containers for liquid scintillation samples) and a background shield have been designed for use with a 50-cm^3 Ge(Li) detector system in laboratory counting of radioactivity in environmental samples (see Figs. III-A, III-B). During counting of the samples, the holder is used to position ten of the sample bottles around the cylindrical surface of the detector, parallel to and symmetric about its axis, and two additional bottles across the end surface of the detector, perpendicular to and symmetric with its axis. With a 500-cm^3 sample and a graded shield developed for use with the system, it is possible to measure 1 pCi/g of ^{232}Th or ^{226}Ra with an error of ± 10% or less and ^{227}Ac within an error of ± 30%.

Pulses are sorted by a 4096-channel analyzer (see Fig. III-C), stored on magnetic tape, and subsequently entered into a computer program which uses an iterative least squares method to identify radionuclides corresponding to those gamma-ray lines found in the sample. The program, which is accessible through a remote terminal, relies on a library of radioisotopes which contains approximately 700 isotopes and 2500 gamma-rays and which runs continuously on the IBM-360 system at ORNL. In identifying and quantifying ^{226}Ra, six principal gamma-ray lines are analyzed. Most of these are from ^{214}Bi and correspond to 295, 352, 609, 1120, 1765, and 2204 KeV. An estimate of the concentration of ^{238}U is obtained from an analysis of the 93 KeV line from its daughter ^{234}Th.

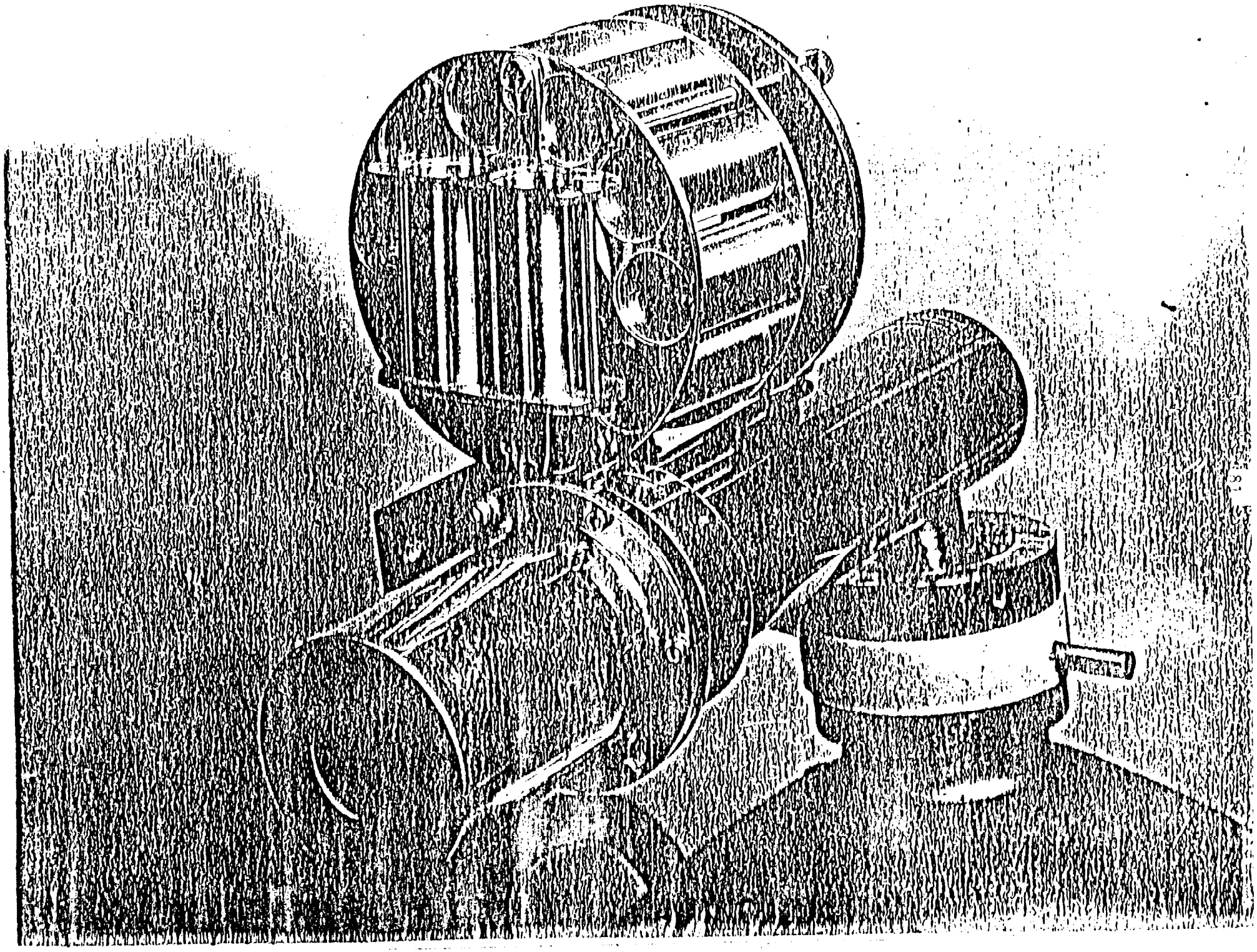
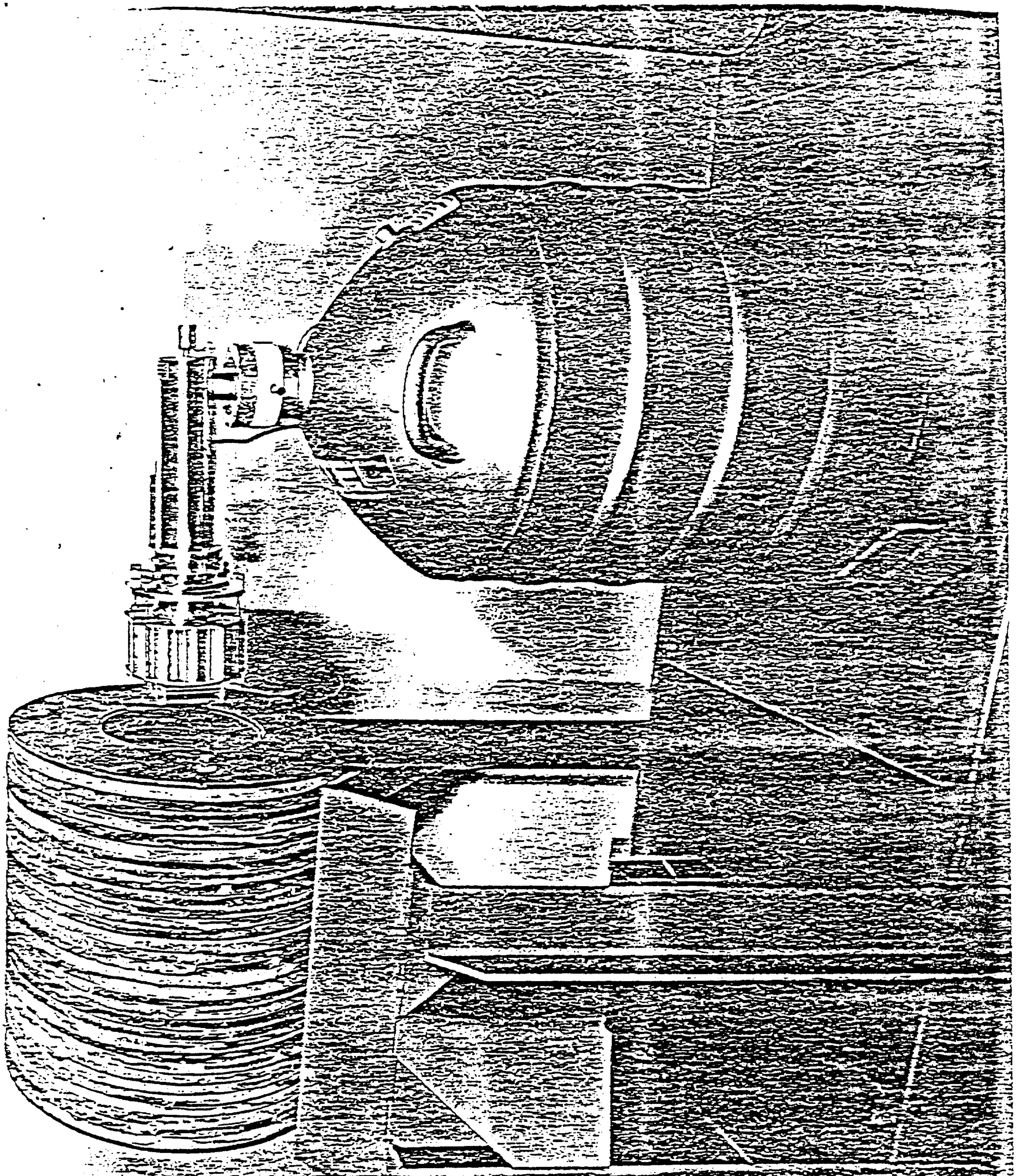


Fig. III-A. Holder for Ge(Li) detector system.



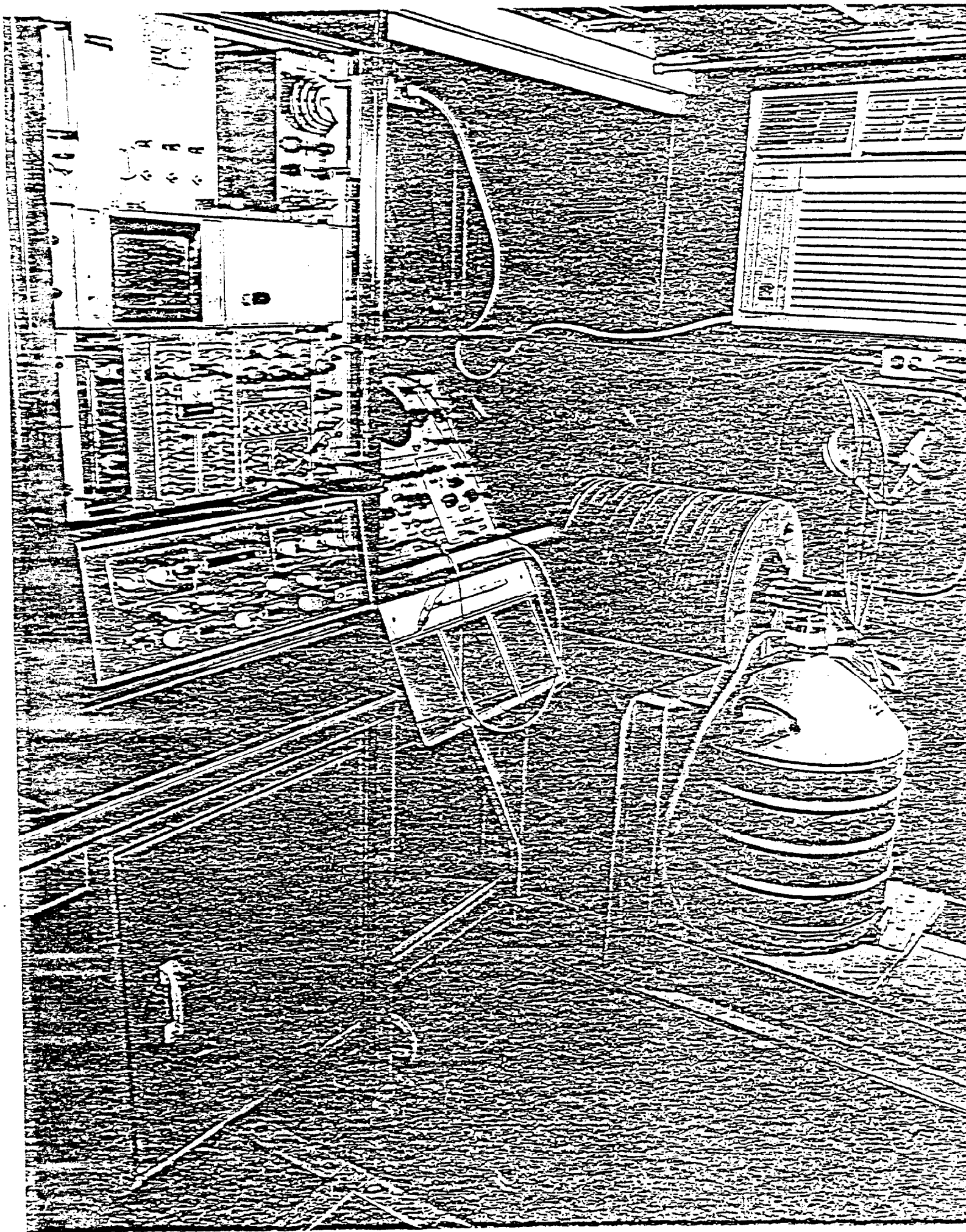


Fig. III-C. 4096-channel analyzer.

APPENDIX IV

PERTINENT RADIOLOGICAL REGULATIONS,

STANDARDS AND GUIDELINES

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

U. S. Nuclear Regulatory Commission
Division of Fuel Cycle and
Material Safety
Washington, D.C. 20555

November 1976

The instructions in this guide in conjunction with Table IV-1 specify the radioactivity and radiation exposure rate limits which should be used in accomplishing the decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table I 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 I prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork 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 ductwork. 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 or 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.

    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 I. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

   a. Identify the premises.

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

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

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

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

TABLE IV-1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[b c f]	MAXIMUM[b d f]	REMOVABLE[b e f]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90 Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except SR-90 and other noted above.	5,000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1,000 dpm $\beta\gamma$/100 cm^2

[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, dpm (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.

Th-nat, Th-232, Sr-90
Ra-223, Ra-224, U-232, I-126,

TABLE IV-1 (cont'd.)

[d] The maximum contamination level applies to an area of not more then 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/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Proposed

ANSI N328-197

Proposed American National Standard

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

Secretariat

Health Physics Society

Property shall not be released for uncontrolled use unless documented measurements show the total and removable contamination levels to be no greater than the values in Table IV-2 or Table IV-3. (Table IV-3 is easier to apply when the contaminants cannot be individually identified.)

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 made the subject of case-by-case evaluation. Credit shall not be taken for coatings over contamination.

TABLE IV-2

SURFACE CONTAMINATION LIMITS

The levels may be averaged* over the 1 m^2 provided the maximum activity in any area of 100 cm^2 is less than 3 times the limit value.

Nuclide	Limit (Activity) dpm/100 cm^2 Total	Removable
Group 1: Nuclides for which the nonoccupational MPC_a** is 2×10^{-13} Ci/m^3 or less or for which the nonoccupational MPC_w*** is 2×10^{-7} Ci/m^3 or less; includes Ac-227; Am-241; -242m, -243; Cf-249; -250, -251, -252; Cm-243, -244, -245, -246, -247, -248; I-125, -129; Np-237; Pa-231; Pb-210; Pu-238, -239 -240, -242, -244; Ra-226, -228; Th-228, -230.****	100	20
Group 2: Those nuclides not in Group 1 for which the nonoccupational MPC_a** is 1×10^{-12} Ci/m^3 or less or for which the nonoccupational MPC_w*** is 1×10^{-6} Ci/m^3 or less; includes Es-254; Fm-256; I-126, -131, -133; Po-210; Ra-223; Sr-90; Th-232; U-232.****	1,000	200
Group 3: Those nuclides not in Group 1 or Group 2.	5,000	1000

*See note following Table 2 on application of limits.

**MPC_a: 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 NCRP., ICRP or NRC (10 CFR Part 20 Appendix B Table 2, Column 1.)

***MPC_w: Maximum Permissible Concentration in Water applicable to members of the public.

****Values presented here are obtained from 10 CFR Part 20. The most limiting of all given MPC values (e.g. soluble vs. insoluble) are to be used. In the event of the occurence of mixture of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fractions must be less than 1.

TABLE IV-3

ALTERNATE SURFACE CONTAMINATION LIMITS

(All alpha emitters, except U-nat and Th-nat are considered as a group)

The levels may be averaged over 1 m^2* provided the maximum activity in any area of 100 cm^2 is less than 3 times the limit value.

Nuclide	Limit (Activity) dpm/100 cm^2 Total	Removable
If the contaminant cannot be identified; or if alpha emitters other than U-nat and Th-nat are present; or if the beta emitters comprise Ac-227, Ra-226, Ra-228, I-125 and I-129.	100	20
If it is known that all alpha emitters are generated from U-nat and Th-nat; and beta emitters are present which, while not identified, do not include Ac-227, I-125, I-129, Ra-226 and Ra-228.	1,000	200
If it is known that alpha emitters are generated only from U-nat and Th-nat; and the beta emitters, while not identified, do not include Ac-227, I-125, I-129, Sr-90, Ra-223, Ra-228, I-126, I-131 and I-133.	5,000	1,000

*NOTE ON APPLICATION OF TABLES 1 AND 2 TO ISOLATED SPOTS OR ACTIVITY:

For purposes of averaging, any m^3 of surface shall be considered to be contaminated above the limit, L, applicable to 100 cm^2 if:

a. From measurements of a representative number, n, of sections, it is determined that $1/n \sum_{i}^{n} Si > L$, where Si is the dpm/100 cm^2 determined from measurement of section i; or

b. On surfaces less than 1 m^2, it is determined that $1/n \sum_{i}^{n} Si \geq AL$, where A is the area of the surface in units of m^2; or

c. It is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3L.

SURGEON GENERAL'S GUIDELINES
Part 712
Grand Junction Remedial Action Criteria

Federal Register, Vol. 41, No. 253, pp. 56777-8, Thursday, December 30, 1976

PART 712 - GRAND JUNCTION REMEDIAL ACTION CRITERIA

712. 1 Purpose

(a) The regulations in this part establish the criteria for determination by ERDA of the need for, priority of and selection of appropriate remedial action to limit the exposure of individuals in the area of Grand Junction, Colo., to radiation emanating from uranium mill tailing which have been used as construction-related material.

(b) The regulations in this part are issued pursuant to Publ. L. 92-514 (86 Stat. 222) of June 16, 1972.

713.2 Scope

The regulations in this part apply to all structures in the area of Grand Junction, Colo., 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, Colo.," means Mesa County, Colo.

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

(d) "ERDA" means the U.S. Energy Research and Development Administration 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 6 air samples, each of at least 100 hours duration, and taken at a minimum of 4-week intervals throughout the year in a habitable area of a structure, or (2) utilizing some other procedure approved by the Commission.

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

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

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

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

(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 mill 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 ERDA other than a written interpretation by the General Counsel will be recognized to be binding upon ERDA.

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. Energy Research and Development Administration, 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 recommend the following graded action levels for remedial action in terms of external gamma radiation level (EGR) and indoor radon daughter concentration level (RDC) above background found within dwellings constructed on or with uranium mill tailings:

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

712.7 Criteria for determination of possible need for remedial action

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

(a) Where ERDA 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 ERDA approved data on indoor radon daughter concentration levels are not available:

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

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

(A) It may be presumed that if the external gamma radiation level is equal to or exceeds 0.02 mR/hr. 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/hr. 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/hr. above background but is less than 0.02 mR/hr. 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/hr. above background averaged on a room-by-room basis.

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

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

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

712.9 Factors to be considered in determination of order or 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.

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

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