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Formerly Utilized MED/AEC Sites Remedial Action Program

Radiological Survey of the Former Virginia-Carolina Chemical Corporation Uranium Recovery Pilot Plant, Nichols, Florida

January 1980

Final Report

Prepared for

U.S. Department of Energy
Assistant Secretary for Environment
Office of Environmental Compliance and Overview
Division of Environmental Control Technology

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PREFACE

This series of reports results from a program initiated in 1974 by the Atomic Energy Commission (AEC) for determination of the condition of sites formerly utilized by the Manhattan Engineer District (MED) and the early 1940's, the control of over 100 sites that were no longer required for nuclear programs has been returned to private industry or the public for unrestricted use. A search of MED and AEC records indicated that for some of these sites, documentation was insufficient to determine whether or not the decontamination work done at the time nuclear activities ceased is adequate by current guidelines.

This report contains the results of a survey of the current radiological condition of the Former Virginia - Carolina Chemical Corporation Uranium Recovery Pilot Plant, Nichols, Florida. Based upon the findings of the survey, there are low levels of radioactivity at various locations at this site and some type of remedial measures should be considered to preclude any future concern of inadvertent radiation exposure to people.

The work reported in this document was conducted by the following members of the Health and Safety Research Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee:

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ABSTRACT

This report presents the results of a radiological survey conducted at the site of a former uranium recovery pilot plant operated by the Virginia-Carolina Chemical Corporation. All that remains of this operation is a concrete pad situated within the boundary of a phosphate products plant now operated by Conserv, Inc., at the Nichols, Florida, site. The survey included measurements designed to characterize the residual radioactivity in the vicinity of this pilot plant and to compare the quantities with federal guidelines for the release of decontaminated property for unrestricted use. The results of this survey indicate that only small quantities of radioactivity exist above normal background levels for that area. Some soil contamination was found in the vicinity of a concrete pad on which the pilot plant stood. Much of this contamination was due to ^{226}Ra and ^{238}U. Some beta-gamma dose rates in excess of applicable guidelines were observed in this same area. External gamma-ray exposure rates at 1 m above the ground range from 20 to 100 μR/hr. None of the direct measurements of alpha contamination were above guideline levels.

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†Consultant.

INTRODUCTION

At the request of the Department of Energy (DOE), a radiological survey was conducted in Nichols, Florida, at the site of a former uranium recovery pilot plant. This plant was designed for the recovery of uranium from wet process phosphoric acid, a regular product of the former Virginia-Carolina Chemical Corporation. This work was performed under contract (March, 1954, amended May 9, 1957) with the Atomic Energy Commission's (AEC) Division of Raw Materials.[1-3] In contract documents the contractor was urged to use "phosphate rock of the highest reasonably available uranium content consistent with the contractor's specifications in its other business for the constituents of the phosphate rock processed." Also, the contract called for the production of all of the U_3O_8 which could be recovered feasibly from the total phosphoric acid production of the plant for five years. Furthermore, the AEC was to purchase over a period of three years, the uranium as a concentrate containing at least 50% U_3O_8. The pilot plant was disassembled shortly after termination of uranium recovery (approximately 1960). The present location of tanks, piping, and other plant equipment used in that operation is unknown. Only a concrete pad remains within the area of an operating phosphoric acid and other phosphate products plant at this site in Nichols, Florida. The present operator of the plant is Conserv, Inc.

On April 4, 1977, H. W. Dickson, Oak Ridge National Laboratory (ORNL), Health and Safety Research Division, and W. T. Thornton, Department of Energy, Division of Safety and Environmental Control, Oak Ridge Operations, met with Mr. Bill Thomas, Conserv representative, for the purpose of conducting a presurvey inspection of the site. Results of exploratory measurements made in the vicinity of a concrete pad on which the pilot plant stood revealed beta-gamma radiation levels at 1 cm above the surface up to 2 mrad/hr and directly measured alpha activity up to 3000 dpm/100 cm^2. Soil samples were taken adjacent to the pad on the north and west sides of a maximum depth of 30 cm. These samples contained approximately 1900 pCi/g of ^{226}Ra and 120 pCi/g of ^{238}U.

Sometime after the presurvey inspection visit, a layer of soil consisting of approximately 3 m^3 was removed from around these locations by the plant operator and was buried about 800 m from the plant in an inactive gypsum pile owned by Conserv. This material was covered with an estimated 0.7 to 1.0 m of gypsum and soil. The area of disposal is marked and was included in this radiological survey conducted by ORNL.

Since the termination of the pilot plant operation, a small building has been constructed on an adjoining concrete pad. The building contains a maintenance shop, lunchroom, tool storage cage, and a small office.

The present survey was undertaken to characterize the existing radiological status of the site. It was conducted by five members of the Health and Safety Research Division, Oak Ridge National Laboratory, and one consultant, on December 12, 13, and 19, 1977.

The survey included: (1) measurement of beta-gamma dose rates at 1 cm and external gamma radiation levels at 1 m above the ground at randomly selected grid points as shown in Fig. 1; (2) direct alpha measurements on the floor, lower walls, ceilings, and roof of a maintenance building covering a portion of the concrete pad and on the surface of the concrete pad which is the location of former pilot plant operations; (3) external gamma-ray measurements at 1 m above ground floor surfaces in areas described in item 2; (4) measurement of ^{226}Ra and ^{238}U in soil samples taken outdoors around the site (see solid circles in Fig. 1) and including an area adjacent to the concrete pad; (5) measurement of ^{226}Ra, ^{238}U, ^{230}Th, and ^{210}Pb in water samples collected on the site; (6) measurement of radon and progeny in air in the building located adjacent to the former pilot plant; and (7) measurement of ^{226}Ra and ^{238}U background in soil samples collected off 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.

Throughout this report the term "lower wall" refers to the surface of a wall up to a height of 2 m, and "overhead surfaces" include wall and ceiling surfaces above 2 m. 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."

SURVEY METHODS

External Gamma Radiation Measurements

A grid pattern was posted on a plan view of an area around the concrete pad. Measurements of the gamma-ray exposure rate were made (1 m above ground) at the intersection of randomly selected lines as indicated by both open and closed circles in Fig. 1. These grid lines are normally site specific and for this survey were spaced approximately 3 m apart. Gamma-ray measurements inside the building were made at a height of 1 m above floor surfaces and in the center of floor areas described by grid markings with a spacing of 2 m. This same procedure was followed in surveying the pad. Results of these measurements were recorded in units of μR/hr.

Two types of instruments were used to measure external gamma radiation levels on this site. One was a scaler equipped Geiger-Mueller (G-M) counter with an energy compensating filter,[4] and the other was a portable NaI scintillation survey meter described in Appendix I. The NaI scintillation survey meter was used to make the gamma measurements reported in this document. The G-M counter operates in an integral mode over a variable period of time, and, hence, is not convenient to use when numerous measurements are required; however, the instrument was used to measure gamma radiation levels at randomly selected locations covering the range of observed NaI survey meter readings, and the recorded NaI readings were then normalized to the G-M counter's response as observed on the site.

Measurements of Beta-Gamma Dose Rates at the Ground Surface

Detailed measurements were made of the beta-gamma dose rates at 1 cm above outside ground (natural, paved, or otherwise covered) surfaces in accordance with the same grid pattern used for external gamma radiation measurements. Open-window and closed-window G-M survey meter readings were made at each indicated grid point or in each survey block. Results of these measurements were recorded in mrad/hr.

The G-M survey 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 for surfaces contaminated with ^{226}Ra in equilibrium with ^{238}U and other nuclides from the uranium chain. At normal background levels, the portable G-M survey meter usually shows readings less than 0.05 mrad/hr (150 cpm) and averaging approximately 0.02 mrad/hr. It should be pointed out that readings within the range of normal background cannot be reproduced accurately on the G-M survey meter. For 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 mrad/hr with an error of ±30%. 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).

Direct Alpha Measurements

Direct measurements of alpha contamination levels were made using alpha scintillation survey meters described in Appendix I. The alpha

survey meters were equipped with scalers which integrate counts over various time intervals, permitting direct alpha measurements on surfaces with low-level alpha contamination. If counts are integrated over a period of 15 sec, the count rate error* associated with a direct reading of 5000 dpm/100 cm^2 (a Nuclear Regulatory Commission [NRC] guideline value for surfaces contaminated with natural uranium) is ±15% for this instrument.

Measurements of Transferable Alpha and Beta Contamination

Transferable alpha and beta contamination levels were measured using the standard smear techniques described in NRC guidelines (Appendix III). The smears were counted using the alpha and beta smear counters shown in Appendix I. The count rate error associated with a 1-min count of a smear showing a transferable alpha contamination level of 1000 cpm/100 cm^2 (an NRC guideline value for natural uranium) is approximately ±6%; the count rate error for a transferable beta contamination level of 1000 dpm/100 cm^2 is approximately ±15%. Indeterminable errors are introduced in taking smear samples because of variations in pressure applied, in surface area actually contacted, and in the condition of the surface.

Investigation of Surface Deposits of Radioactivity

Samples of soil were collected on the surface and beneath the surface to a maximum depth of 75 cm, around the concrete pad at points shown in Fig. 1 as solid circles. The locations (and designations) of these samples correspond to grid points where surface beta-gamma measurements were made. In addition, soil and water samples were collected at other points on the site in order to establish an on-site value for uranium and radium background.

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

Radon Measurements

Because of the high concentration of radium (1900 pCi/g) found in soil near the concrete pad, a decision was made to measure radon and radon daughter concentrations in the building. Radon measurements were taken using a continuous radon monitor of the Wrenn type.[5] For purposes of comparison, one detector head of the unit was placed inside the lunchroom and one was placed outside in the more open maintenance area. The Wrenn chamber, described in Appendix I, was attached to a printer which recorded automatically data proportional to the radon concentration 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 the maintenance and lunchroom areas, air was pumped for 10 min at approximately 18 liters/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 I.

Methods Used to Analyze Samples

Soil samples collected at outdoor grid points were packed in plastic bags and returned to ORNL, where they were dried for 24 hr at 110°C and then pulverized to a particle size no greater than 500 μm in diameter (-35 mesh). Next, 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 technique is given in Appendix II. Concentrations of ^{226}Ra, ^{238}U, and ^{227}Ac were estimated in this way. In addition, a second measurement of the ^{238}U concentration in each sample was obtained by neutron absorption techniques.[6] Also analyzed for ^{238}U by this technique were residue samples collected from each survey grid block on the concrete pad.

Water samples collected on the site and removed from drains were analyzed by the Analytical Chemistry Division of ORNL for ^{210}Pb, ^{226}Ra, and ^{230}Th, using techniques described in appendices to the ORNL Master Manual. The concentration of ^{238}U in water samples was determined using the above neutron absorption technique.

Background Measurements

Background external gamma radiation levels at 1 m above the ground were measured at points several miles from the site in each compass direction (see locations FL3, FL4, FL5, FL6, and FL7 in Fig. 2). The average of these background measurements was 5 μR/hr. Soil samples taken at these 5 points showed ^{226}Ra concentrations ranging from 0.3 to 2.3 pCi/g, ^{238}U concentrations ranging from 0.1 to 1.1 pCi/g, and ^{232}Th concentrations ranging from 0.3 to 0.4 pCi/g. The concentration of ^{227}Ac was below the detectable level in all samples. Two additional measurements of the external gamma radiation level at 1 m above the ground were made at locations within the confines of the plant but sufficiently removed from the surveyed area so as to be unaffected by the former operations. The average of these measurements was 41 μR/hr, indicating widespread elevation of the gamma exposure rate, presumably due to present plant operations where uranium-bearing phosphate rock is handled.

Background beta-gamma dose rates, as measured with the G-M survey meters used on the site, are typically in the range of 0.01 to 0.05 mrad/hr. Geiger-Mueller meter readings in this range (below 100 cpm) are not generally reproducible; however, these typical background readings are reported for purposes of comparison with on-site readings. Beta-gamma doses were measured at the two locations mentioned in the previous paragraph and averaged approximately 0.07 mrad/hr. Background alpha readings for the type of alpha survey meter used on this site are negligible.

All direct survey 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. For the measurements 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.

SURVEY RESULTS

Measurements of Alpha and Beta-Gamma Contamination and External Gamma Radiation Levels in the Maintenance Building

This building, which measures 10 x 17 m, contains a maintenance room and tool cage (open on the north side), a lunchroom, and a small office. The inside dimensions of the maintenance room were 9.8 x 11.4 m. Dimensions of the lunchroom were 5.4 x 6.8 m and the small office measured 2.8 x 5.4 m. The ceilings over the lunchroom and office were 2.4 m high and finished with wood paneling. The ceiling over the maintenance room was open steel girder construction and varied in height from a minimum of about 2.4 m to a maximum of about 3.6 m, sloping downward from north to south. Measurements were made in accordance with survey methods described earlier. These measurements were made in survey blocks as shown in Fig. 3.

Directly measured alpha levels, averaged over individual survey blocks on the maintenance room floor, varied from 80 cpm/100 cm^2 to 310 dpm/100 cm^2 (see Table 1). The maximum observed individual alpha measurement was 420 dpm/100 cm^2. Direct beta-gamma dose rates at 1 cm from the floor surface of the maintenance room varied from 0.04 mrad/hr to 0.07 mrad/hr and are given in Table 1. Transferable alpha and beta-gamma contamination levels on the floor were negligible. External gamma radiation levels at 1 m above the floor ranged from 8 μR/hr to 20 μR/hr. Results of direct alpha and beta-gamma readings on overhead surfaces of the ceiling of the maintenance room are given in Table 2.

Results of direct alpha, beta-gamma, and external gamma radiation for the lunchroom floor and lower walls are given in Table 3. No transferable alpha or beta-gamma contamination was observed in this area. No readings were taken on the walls of the office since nearly all surfaces

were inaccessible. Additionally, since the ceilings of the lunchroom and office were relatively new construction, no measurements were made on these surfaces.

Measurements of Alpha and Beta-Gamma Contamination Levels on Building Roof

Results of direct alpha and beta-gamma contamination levels for the roof of the building are given in Table 4. The survey grid used for the roof is given in Fig. 4. Alpha readings ranged from 180 dpm/100 cm^2 to 400 dpm/100 cm^2. Direct beta-gamma dose rates at 1 cm from the roof surface ranged from 0.03 mrad/hr to 0.08 mrad/hr. No transferable alpha or beta-gamma contamination was observed on the roof.

Measurements of Alpha and Beta-Gamma Contamination and External Gamma Radiation Levels on the Concrete Pad

The concrete pad located at the west end of the maintenance building, described previously, measured approximately 10 x 10 m. This area was bare except for some stored material and a layer of soil. The pad was divided into survey blocks as shown in Fig. 5. Results of direct alpha and beta-gamma surface measurements and external gamma radiation levels at 1 m above the surface of the pad are given in Table 5. The maximum and average direct alpha and beta-gamma measurements in each survey block are shown in Fig. 5. Directly measured alpha and beta-gamma measurements were made at randomly selected spots within each survey block on the pad. Averages of direct alpha measurements within survey blocks ranged from 200 dpm/100 cm^2 to 1400 dpm/100 cm^2. The maximum observed measurement was 2900 dpm/100 cm^2. Averages of beta-gamma dose rates within survey blocks ranged from 0.05 mrad/hr to 0.26 mrad/hr (survey block B4). The maximum observed individual beta-gamma measurement was 0.5 mrad/hr. In two survey blocks the average beta-gamma dose rate exceeded the NRC guide value of 0.2 mrad/hr (see Fig. 5). External gamma-ray exposure rates at 1 m ranged from 22 to 58 μR/hr. Because the concrete pad surface was covered with a layer of residue (ranging from a few millimeters to 3 cm thick), standard smear samples were not taken. Instead, samples of the residue were scraped from the surface and analyzed

for ^{238}U using neutron activation techniques.[6] The results show that ^{238}U concentrations ranged from 15 pCi/g to 56 pCi/g. The data are presented in Table 6. Soil removed from a floor drain found in survey block E5 showed a concentration of 63 pCi/g of ^{238}U and 44 pCi/g of ^{226}Ra.

Measurements of Beta-Gamma and External Gamma Radiation Levels at Outdoor Locations

Measurements of beta-gamma dose rates at 1 cm above the ground at circled grid points in Fig. 1 and external gamma radiation levels at 1 m above the ground at these grid points outdoors are given in Table 7. Beta-gamma dose rates over the area averaged 0.09 mrad/hr. Dose rates exceeded the average NRC guide value of 0.20 mrad at two grid points (see Table 7). These grid points are in the larger area indicated by shading in Fig. 6, where beta-gamma dose rates were generally in the range of 0.20 to 1.0 mrad/hr. At outdoor grid points shown in Fig. 1, gamma-ray exposure rates ranged from 20 to 100 μR/hr as shown in Table 7. The average of all gamma-ray exposure rates in Table 7 is 55 μR/hr.

Results of Soil Sample Analyses

Surface soil samples were taken on the site at the solid circles shown in Fig. 1. These samples contained up to 54 pCi/g ^{238}U and 950 pCi/g ^{226}Ra. Results are given in Table 8. Core samples were taken at four locations adjacent to the concrete pad at selected grid points. These samples were taken at depths to 75 cm using a hand-operated, split-spoon sampler. Results of these core samples are given in Table 9. It is seen that the uranium in these samples was approximately the same as that in surface samples. Also, it is seen that the major ^{226}Ra contamination in soil is found in the upper 23 cm.

Results of Water Sample Analyses

Water samples were taken at three locations on the site. The maximum concentration of uranium (110 pCi/liter) was found in a sample

from a small stream of water draining from a portion of the plant now in operation (grid point A12, Fig. 1). A water sample taken from standing water located at grid point U6 showed a ^{238}U concentration of less than 3 pCi/liter. A similar figure was obtained from a water sample, taken from a faucet in the toilet building indicated by dashed lines south of the concrete pad shown in Fig. 1.

External Gamma Radiation Levels 1 m Above the Ground at the Site of Disposal of Contaminated Soil Removed from the Vicinity of the Former Pilot Plant

External gamma radiation levels at 1 m from the ground were measured at 9 uniformly separated points in a 6 x 6 m area containing dirt which had been removed from the vicinity of the pad and deposited on a gypsum pile approximately 800 yards south of the plant's east gate. The measurements ranged from 20 to 40 μR/hr and averaged 23 μR/hr.

One surface and two subsurface soil samples were taken from this area, at points showing highest gamma radiation levels. The samples showed ^{226}Ra concentrations ranging from 5 to 35 pCi/g and ^{238}U concentrations ranging from 2 to 8 pCi/g (see Table 9).

Radon and Radon Daughter Measurements Inside Lunchroom and Maintenance Area

During the period December 12 through 14, 1977, the radon concentration in air was measured continuously in the lunchroom and in the more open maintenance area. Results of analyses of these samples indicate that in the lunchroom, the ^{222}Rn concentration ranged from 0.4 pCi/liter to 1.0 pCi/liter with an average of 0.6 pCi/liter. In the open maintenance area, the concentration ranged from 0.2 to 1.9 pCi/liter with an average of 0.9 pCi/liter. Additionally, air samples were collected and analyzed for radon daughters on December 12. Analyses of these data indicate only trivial radon daughter concentrations.

SUMMARY

The maintenance building at this site showed low levels of contamination. Measurements of direct alpha and beta-gamma contamination

revealed values below guidelines set by the NRC, and external gamma radiation levels at 1 m were below 20 μR/hr. Transferable alpha and beta-gamma contamination was negligible both on the floors and walls and on overhead surfaces. Radon levels in the building averaged 0.9 pCi/liter, well below the value of 3 pCi/liter given in 10 CFR 20 Appendix B, set for nonoccupational exposure. Radon daughter levels were negligible. The roof showed very little contamination with direct alpha and beta-gamma measurements being well below NRC guidelines with no transferable contamination being found.

Direct alpha contamination measurements on a concrete pad (site of the former pilot plant) show values below NRC guidelines for surfaces contaminated with uranium. Beta-gamma dose rates were also below the level of 0.2 mrad/hr as set by the NRC with the exception of the two points shown in Fig. 2. External gamma exposure rates at 1 m were all below 60 μR/hr. Samples of residue from the pad were analyzed and show low levels of uranium concentration.

Measurements of beta-gamma and external gamma radiation levels in the area surrounding the concrete pad and building give averages of 0.09 mrad/hr and 55 μR/hr, respectively. Two points show beta-gamma dose rates which exceed the limit of 0.2 mrad/hr. Surface soil samples which were taken in the area showing highest beta-gamma dose rates show concentrations of as high as 54 pCi/g ^{238}U and 950 pCi/g ^{226}Ra. Three water samples were taken and showed a maximum uranium concentration of 110 pCi/liter. This concentration was found in water from a small stream draining from a portion of the plant now in operation. Water from other sources (i.e., standing surface water and water from a potable tap) contained negligible uranium.

A small area containing soil which had been removed from around the concrete pad showed external gamma levels at 1 m of 20 to 40 μR/hr. Soil samples taken at this site contained less than 35 pCi/g of ^{226}Ra and less than 8 pCi/g of ^{238}U.

An evaluation has been made of current radiation exposures at this site and is presented in Appendix IV (page 75) of this report. The purpose of this evaluation is to present information which will permit the reader to compare current radiation exposures from the site to

normal background exposures for that part of Florida, as well as to scientifically based guideline values established for the protection of radiation workers and members of the general public.

REFERENCES

1. Division of Raw Materials, U.S. Atomic Energy Commission, Letter Contract No. AT(49-1)-623 with Virginia-Carolina Chemical Corporation (May 23, 1952).
2. Division of Raw Materials, U.S. Atomic Energy Commission, Contract No. AT(49-6)-623 with Virginia-Carolina Chemical Corporation (March 17, 1954).
3. Division of Raw Materials, U.S. Atomic Energy Commission, Contract No. AT(49-6)-623 Amendment No. 1 (May 9, 1957).
4. G. S. Hurst and E. B. Wagner, "A Geiger-Mueller Gamma Ray Dosimeter with Low Neutron Sensitivity," *Health Phys.* 5, Nos. 1/2, pp. 20-26 (April 1961).
5. M. E. Wrenn, H. Spitz, and N. Cohen, "Design of a Continuous Digital Output Environmental Radon Monitor," *IEEE Trans. Nucl. Sci.* 22, 645 (1975).
6. F. F. Dyer, J. F. Emery, and G. W. Leddicotte, *Comprehensive Study of the Neutron Activation Analysis of Uranium by Delayed Neutron Counting*, Oak Ridge National Laboratory Report ORNL-3342 (October 1962).

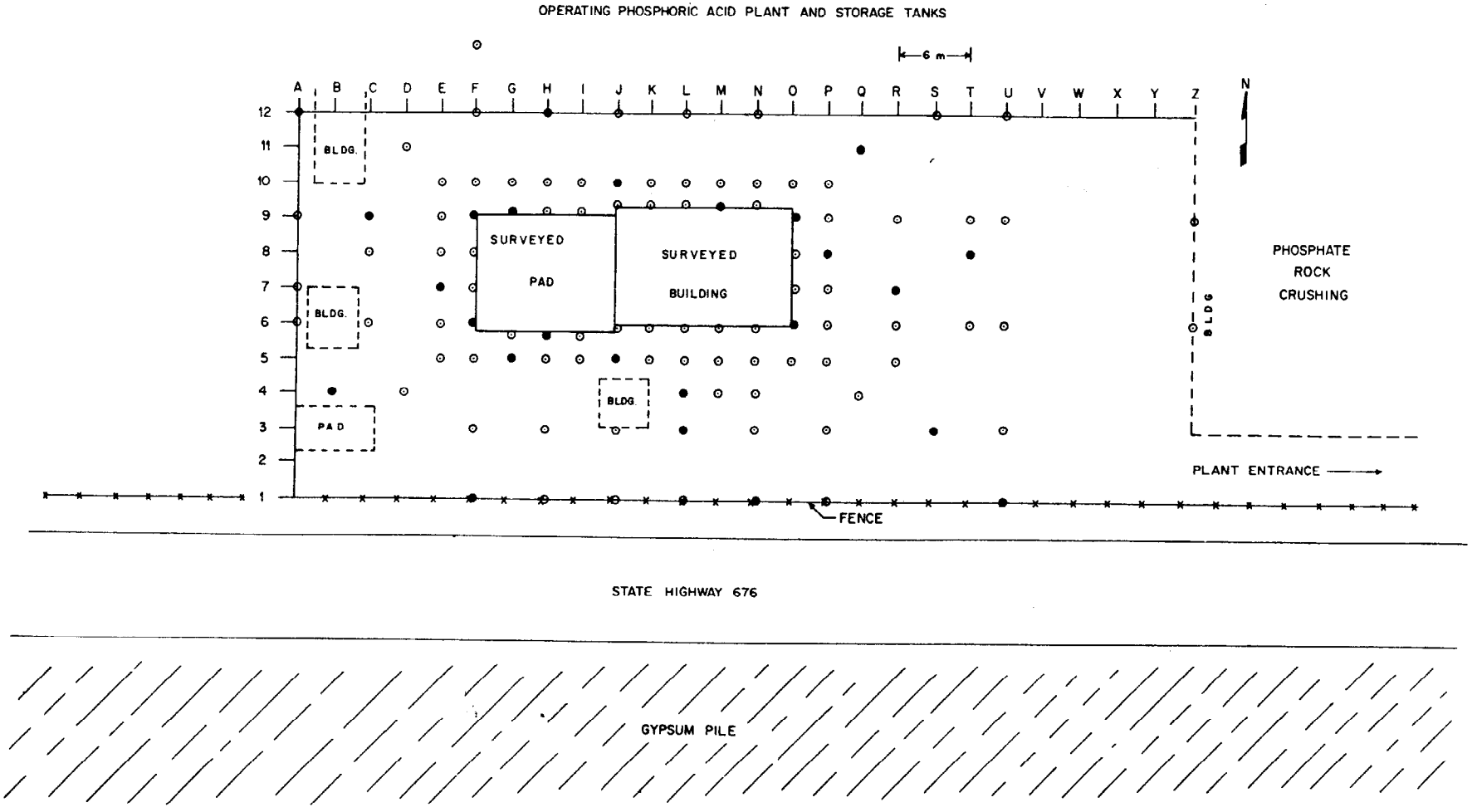

Fig. 1. Plan view of survey site showing grid system (darkened circles are both soil sampling points and meter measurement points; other circles are meter measurement points).

Fig. 2. Background sampling points (FL3 through FL7) in the Nichols area.

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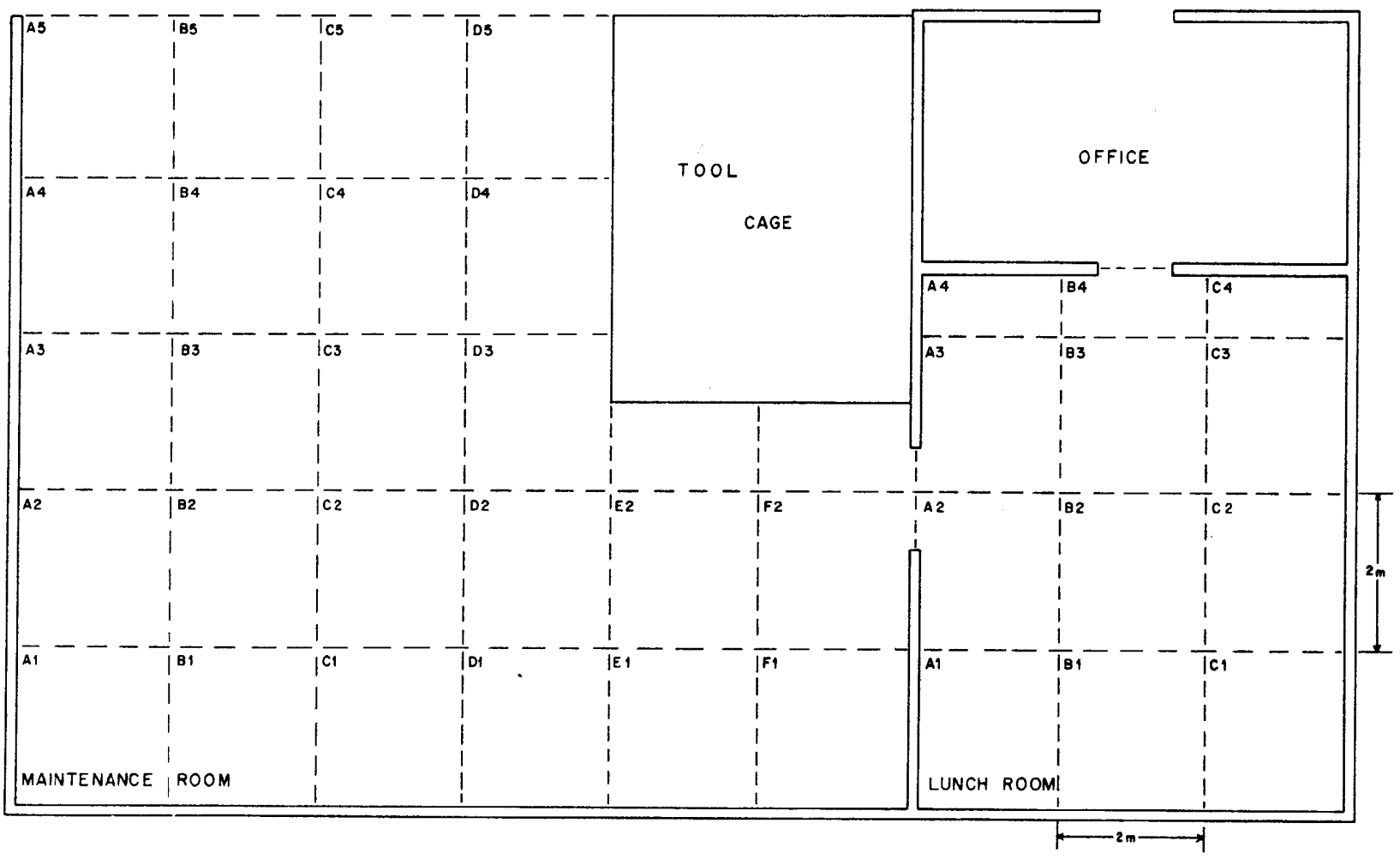

Fig. 3. Plan view of maintenance building showing survey blocks.

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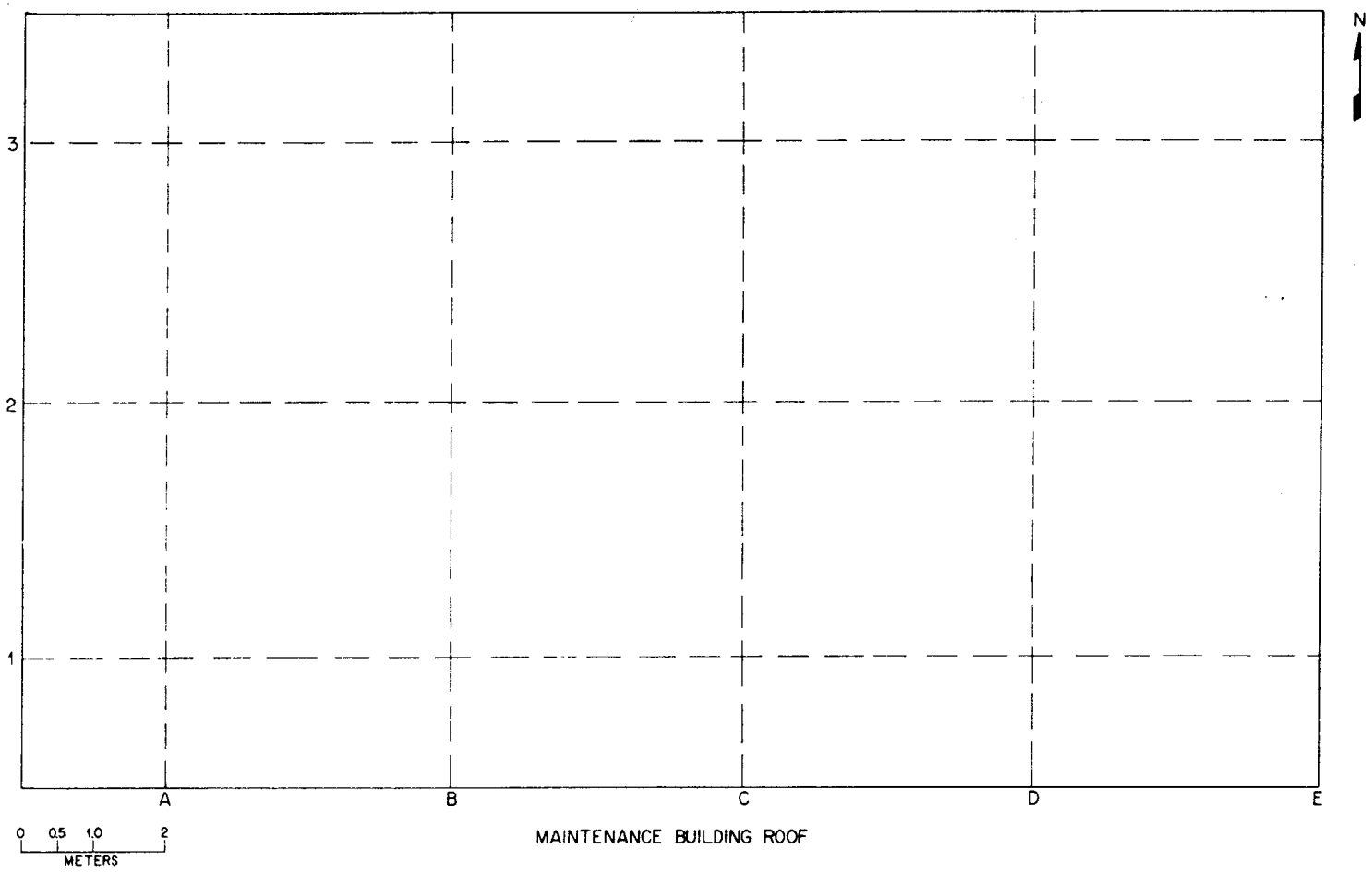

Fig. 4. Grid system used on roof of maintenance building.

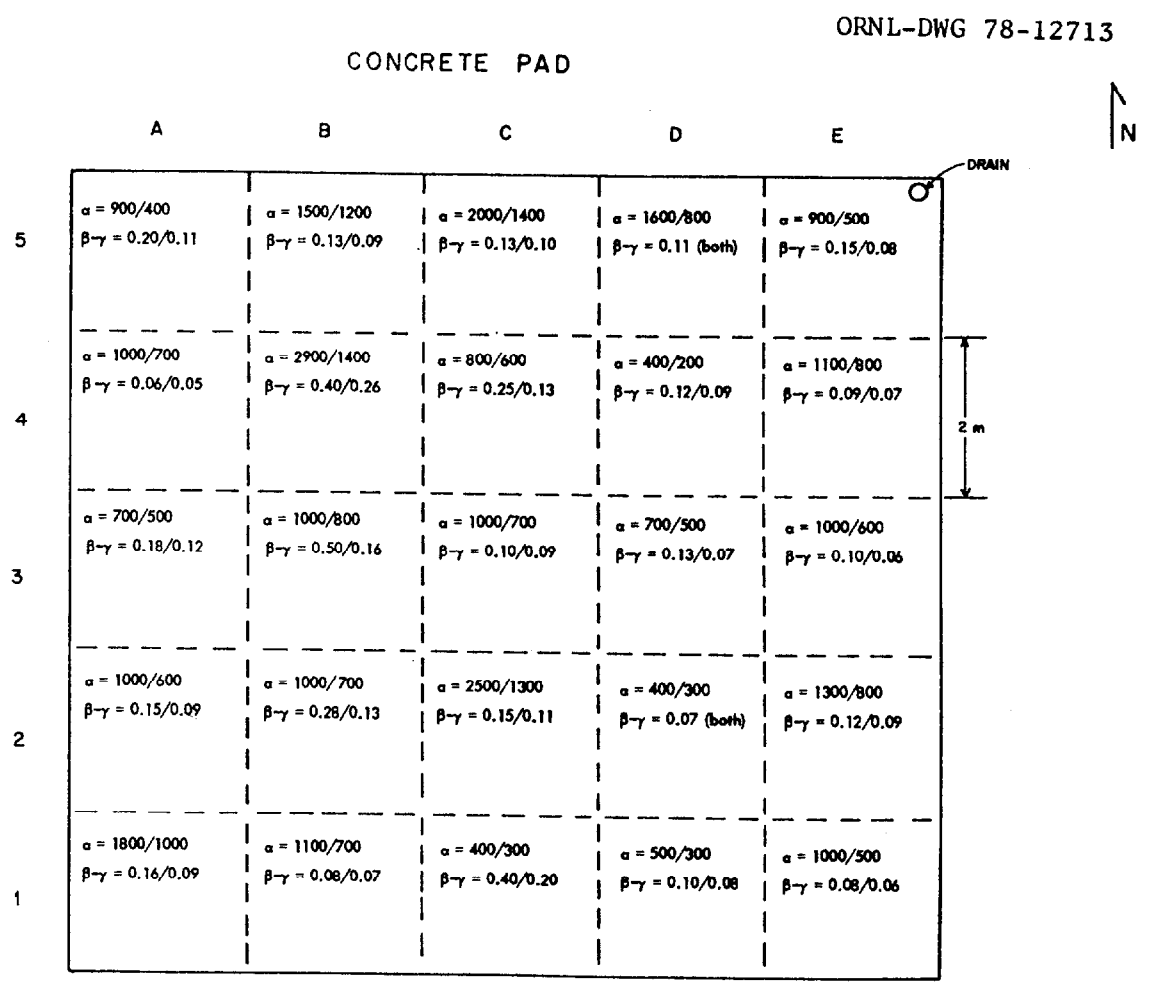

Fig. 5. Survey blocks used on concrete pad with maximum/average direct measurements for alpha (dpm/100 cm^2) and beta-gamma (mrad/hr) in each block.

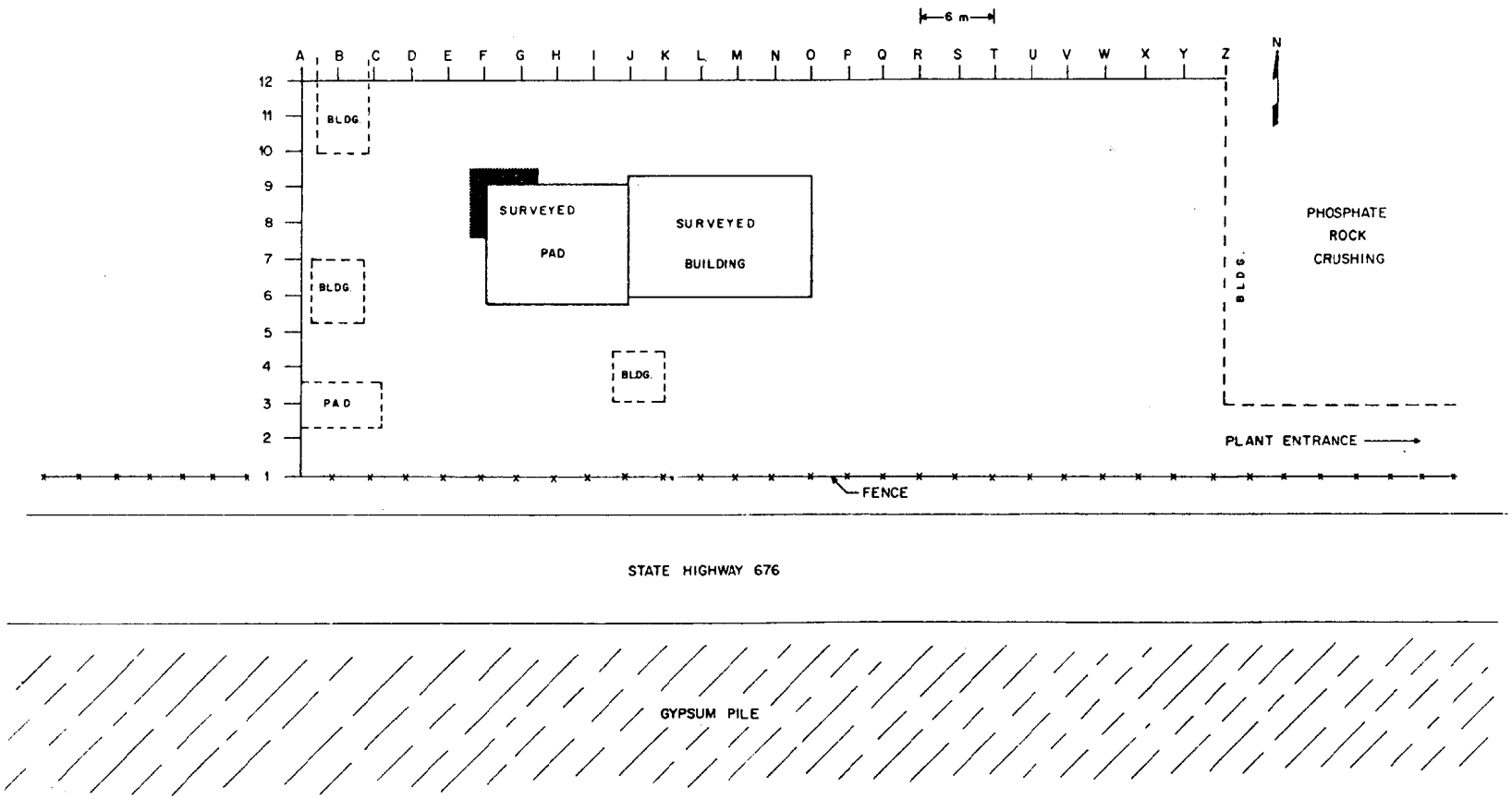

Fig. 6. Area near concrete pad showing beta-gamma dose rates in excess of NRC guidelines (see shaded area).

Table 1. Directly measured alpha and beta-gamma contamination levels and external gamma radiation levels in survey blocks in maintenance room

Survey block shown in Fig. 3	Direct alpha measurements averaged over 1 m^2 (dpm/100 cm^2)	Maximum observed direct alpha measurements (dpm/100 cm^2)	Beta-gamma dose rate at 1 cm averaged over 1 m^2 (mrad/hr)	Maximum observed beta-gamma dose rate at 1 cm (mrad/hr)	External gamma radiation level at 1 m (μR/hr)
A1	150	210	0.05	0.07	12
A2	230	310	0.05	0.06	12
A3	80	210	0.04	0.05	12
A4	150	210	0.05	0.06	12
A5	190	310	0.06	0.06	20
B1	150	210	0.05	0.06	10
B2	190	420	0.05	0.07	12
B3	170	210	0.05	0.05	12
B4	170	310	0.05	0.06	12
B5	80	210	0.07	0.08	20
C1	310	310	0.04	0.04	10
C2	100	210	0.04	0.05	11
C3	230	310	0.05	0.05	12
C4	120	210	0.06	0.08	12
C5	230	420	0.06	0.07	15
D1	100	210	0.05	0.06	10
D2	230	310	0.05	0.05	10
D3	170	310	0.06	0.06	10
D4	190	310	0.05	0.07	12
D5	190	310	0.06	0.07	16
E1	170	310	0.05	0.06	11
E2	230	420	0.05	0.06	10
F1	100	210	0.04	0.05	11
F2	80	210	0.04	0.05	11
Tool cage	170	310	0.05	0.06	8

Table 2. Directly measured alpha and beta-gamma contamination levels on overhead surfaces in maintenance room

Survey block shown in Fig. 3	Vertical surfaces		Horizontal surfaces	
	Alpha contamination level (dpm/100 cm^2)	Beta-gamma dose rate (mrad/hr)	Alpha contamination level (dpm/100 cm^2)	Beta-gamma dose rate (mrad/hr)
A1	0	0.04	0	0.04
A2	0	0.04	0	0.04
A3	0	0.05	0	0.05
A4	0	0.04	0	0.06
A5	0	0.06	0	0.06
B1	0	0.04	0	0.04
B2	0	0.04	0	0.04
B3	0	0.05	0	0.05
B4	0	0.04	0	0.06
B5	0	0.05	0	0.05
C1	0	0.03	0	0.03
C2	0	0.03	0	0.03
C3	0	0.03	0	0.03
C4	0	0.05	0	0.05
C5	0	0.05	0	0.05
D1	0	0.03	0	0.03
D2	0	0.03	0	0.04
D3	0	0.03	0	0.03
D4	0	0.05	0	0.05
D5	0	0.05	0	0.05
E1	0	0.04	0	0.04
E2	0	0.04	0	0.04
F1	0	0.04	0	0.04
F2	0	0.04	0	0.04
Tool cage				
A6	100	0.05	100	0.06
B6	100	0.06	0	0.06
C6	0	0.05	0	0.08
D6	0	0.05	0	0.04
E6	0	0.05	0	0.06
F6	0	0.06	0	0.05

Table 3. Directly measured alpha and beta-gamma contamination levels and external gamma radiation levels in survey blocks in lunchroom

Survey block shown in Fig. 3	Direct alpha measurements averaged over 1 m^2 (dpm/100 cm^2)	Maximum observed direct alpha measurements (dpm/100 cm^2)	Beta-gamma dose rate at 1 cm averaged over 1 m^2 (mrad/hr)	Maximum observed beta-gamma dose rate at 1 cm (mrad/hr)	External gamma radiation level at 1 m (μR/hr)
A1	40	100	0.04	0.05	18
A2	80	200	0.04	0.05	18
A3	40	100	0.04	0.06	18
A4	80	200	0.04	0.05	16
B1	140	200	0.03	0.04	18
B2	60	200	0.04	0.05	18
B3	180	300	0.04	0.04	20
B4	20	100	0.04	0.04	18
C1	40	100	0.03	0.04	16
C2	120	200	0.03	0.04	18
C3	100	200	0.04	0.05	16
C4	60	100	0.03	0.04	16
Walls					
A1W	20	100	0.03	0.04	NA[a]
A2W	0	0	0.03	0.03	NA
A3W	0	0	0.02	0.03	NA
A4W	0	0	0.02	0.03	NA
A4N	40	100	0.03	0.03	NA
B4N	0	0	0.02	0.03	NA
C4N	0	0	0.02	0.03	NA
C4E	20	100	0.03	0.04	NA
C3E	0	0	0.03	0.04	NA
C2E	0	0	0.03	0.03	NA
C1E	0	0	0.03	0.03	NA
C1S	0	0	0.03	0.04	NA
B1S	20	100	0.02	0.03	NA
A1S	0	0	0.02	0.03	NA

[a] NA = Not applicable for lower walls.

Table 4. Direct measurements of alpha and beta-gamma contamination on roof of building

Survey block	Beta-gamma dose rate (mrad/hr)	Alpha contamination level (dpm/100 cm^2)
A1	0.04	200
B1	0.03	200
C1	0.05	200
D1	0.05	200
E1	0.06	200
A2	0.04	300
B2	0.06	200
C2	0.05	200
D2	0.05	200
E2	0.05	200
A3	0.07	300
B3	0.08	200
C3	0.05	180
D3	0.07	100
E3	0.08	400

Table 5. Directly measured alpha and beta-gamma contamination levels and external gamma radiation levels in survey blocks on concrete pad

Survey block shown in Fig. 5	Direct alpha measurements averaged over 1 m^2 (dpm/100 cm^2)	Maximum observed direct alpha measurements (dpm/100 cm^2)	Beta-gamma dose rate at 1 cm averaged over 1 m^2 (mrad/hr)	Maximum observed beta-gamma dose rate at 1 cm (mrad/hr)	External gamma radiation level at 1 m (μR/hr)
A1	1,000	1,800	0.09	0.16	52
A2	600	1,000	0.09	0.15	42
A3	500	700	0.12	0.18	42
A4	700	1,000	0.05	0.06	44
A5	400	900	0.11	0.20	58
B1	700	1,100	0.07	0.08	44
B2	700	1,000	0.13	0.28	38
B3	800	1,000	0.16	0.50	36
B4	1,400	2,900	0.26	0.40	38
B5	1,200	1,500	0.09	0.13	44
C1	300	400	0.20	0.40	42
C2	1,300	2,500	0.11	0.15	30
C3	700	1,000	0.09	0.10	28
C4	600	800	0.15	0.25	34
C5	1,400	2,000	0.10	0.13	40
D1	300	500	0.08	0.10	36
D2	300	400	0.07	0.07	28
D3	500	700	0.07	0.13	22
D4	200	400	0.09	0.12	28
D5	800	1,600	0.11	0.11	32
E1	500	1,000	0.06	0.08	28
E2	800	1,300	0.09	0.12	26
E3	600	1,000	0.06	0.10	22
E4	800	1,100	0.07	0.09	26
E5	500	900	0.08	0.15	32

Table 6. Uranium-238 concentrations in surface residue taken from the concrete pad

Grid location shown in Fig. 5	^{238}U concentration (pCi/g)
A1	42
A2	38
A3	33
A4	15
A5	28
B1	48
B2	45
B3	43
B4	17
B5	31
C1	44
C2	42
C3	45
C4	18
C5	21
D1	44
D2	52
D3	41
D4	45
D5	56
E1	41
E2	52
E3	36
E4	50
E5	45
E5[a]	63

[a]Drain sample from northeast corner; this sample contained 44 pCi/g of ^{226}Ra.

Table 7. External gamma radiation levels and beta-gamma dose rates at grid points outdoors on the site

Grid point shown in Fig. 1	Beta-gamma dose rate at 1 cm (mrad/hr)	External gamma radiation level at the surface (μR/hr)	External gamma radiation level at 1 m (μR/hr)
A6	0.06	63	42
A7	0.08	69	56
A9	0.10	63	64
A12	0.05	31	32
B4	0.06	38	40
C6	0.07	47	56
C8	0.14	94	70
C9	0.10	69	62
D4	0.10	88	66
D11	0.07	50	46
E5	0.12	88	88
E6	0.10	94	82
E7	0.09	94	90
E8	0.18	160	100
E9	0.13	63	72
E10	0.08	78	66
F1	0.10	94	56
F3	0.11	69	66
F5	0.10	78	80
F6	0.13	94	96
F7	0.15	130	100
F8	0.18	160	86
F9	0.75	800	100
F10	0.09	78	68
F12	0.06	63	50
F14	0.09	78	48
G5	0.13	130	90
G6	0.12	88	76
G9	0.22	200	68
G10	0.09	81	68
H1	0.08	78	48
H3	0.10	63	70
H5	0.09	94	70
H6	0.08	63	54
H9	0.18	140	56
H10	0.06	47	56
H12	0.12	94	48
I5	0.10	69	54
I6	0.08	63	54
I9	0.04	25	40
I10	0.08	63	48
J1	0.06	47	42
J3	0.08	69	46

Table 7 (cont.). External gamma radiation levels and beta-gamma dose rates at grid points outdoors on the site

Grid point shown in Fig. 1	Beta-gamma dose rate at 1 cm (mrad/hr)	External gamma radiation level at the surface (μR/hr)	External gamma radiation level at 1 m (μR/hr)
J5	0.06	63	50
J6	0.10	63	42
J9	0.03	31	28
J10	0.08	78	48
J12	0.04	25	20
K5	0.06	63	46
K6	0.06	56	40
K9	--	--	20
K10	0.06	63	42
L1	0.06	63	40
L3	0.10	63	50
L5	0.06	63	48
L6	0.06	47	34
L9	0.04	16	32
L10	0.08	63	48
L12	0.03	31	24
M5	0.10	78	54
M6	0.08	78	50
M9	0.03	31	28
M10	0.08	78	52
N1	0.09	88	50
N3	0.10	94	62
N5	0.11	100	62
N6	0.10	88	44
N9	0.02	16	28
N10	0.10	88	50
N12	0.03	25	28
O5	0.12	93	62
O6	0.09	94	58
O7	0.08	31	34
O8	0.07	47	38
O9	0.07	56	38
O10	0.10	88	56
P1	0.08	69	46
P3	0.09	94	58
P5	0.09	88	66
P6	0.09	100	64
P7	0.12	63	64
P8	0.08	78	60
P9	0.08	69	62
P10	0.08	78	64
Q4	0.09	78	62
Q11	0.14	100	66

Table 7 (cont.). External gamma radiation levels and beta-gamma dose rates at grid points outdoors on the site

Grid point shown in Fig. 1	Beta-gamma dose rate at 1 cm (mrad/hr)	External gamma radiation level at the surface (μR/hr)	External gamma radiation level at 1 m (μR/hr)
R6	0.10	78	64
R7	0.09	88	64
R9	0.09	78	64
S3	0.09	75	56
S12	0.05	31	28
T6	0.10	81	54
T8	0.08	63	56
T9	0.07	69	54
U1	0.06	56	48
U3	0.08	69	52
U6	0.08	69	56
U9	0.08	69	56
U12	0.03	31	46
Z6	0.10	78	60
Z9	0.08	78	56
L4	0.08	69	36
M4	0.07	47	32
N4	0.10	88	46

Table 8. Radionuclide concentrations in surface soil samples

Grid location shown in Fig. 1	^{238}U (pCi/g)	^{226}Ra (pCi/g)
A12	29	27
B4	6	22
C9	29	42
E7	35	55
F1	53	55
F6	41	290
F9	26	950
G9	30	520
G5	40	55
H6	54	59
H12	28	83
J5	32	31
J10	37	45
L3	46	46
L4	47	48
M9 (+0.7 m)	47	40
N1	43	44
O6	53	48
O9	32	35
P8	50	50
Q11	28	44
R7	50	48
S3	42	42
T8	43	40
125N	20	14
125W	8	41
None[a]	2	1

[a]Off-site sample taken 0.85 mile west of survey area.

Table 9. Radionuclide concentrations in subsurface soil samples

Grid location shown in Fig. 1	Depth (cm)	^{238}U (pCi/g)	^{226}Ra (pCi/g)
F8	0 - 23	13	240
F8	23 - 46	4	42
F9	0 - 23	46	290
F9	23 - 46	31	31
F9A	0 - 23	24	360
F9A	23 - 46	22	19
G5	0 - 23	35	83
G5	23 - 46	23	29
G5	46 - 76	4	6
H9	0 - 23	42	240
H9	23 - 46	29	37
None[a]	surface	3	5
	0 - 46	2	6
	46 - 76	8	35

[a]Core sample collected from the center of a 6 x 6 m area of an inactive gypsum pile, the burial site for approximately 3 m^3 of contaminated soil removed from the north and west sides of the concrete pad shown in Fig. 1.

APPENDIX I

DESCRIPTION OF RADIATION SURVEY INSTRUMENTS

RADIATION SURVEY METERS

Alpha Survey Meters

The type of alpha survey meter used 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 (100 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 0.15-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-Mueller (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 (see 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 an NBS standard ^{226}Ra source, was 3200 cpm per mR/hr.

Gamma Scintillation Survey Meter

A portable survey meter using a sodium iodide (NaI) scintillation probe is used to measure low-level gamma radiation exposure. The scintillation probe consists of 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 exposure rate conversion factors are of the order of 300 cpm per μR/hr. The sensitivity of this instrument may be influenced by factors such as temperature, humidity, and small changes in photomultiplier tube voltage. Therefore, each instrument used in the field is standardized daily, and its response is compared with readings made with a gamma-ray dosimeter developed at ORNL by Hurst and Wagner[I-1] and called the "Phil" dosimeter. This latter instrument, described below, has response which is proportional to exposure in Roentgens over a wide energy range. Readings made with the portable scintillation survey meter and compared with exposure rates determined at the same time using the "Phil" may be used as a factor to convert the reading in counts per unit time to exposure rate per unit time (μR/hr).

"Phil" Gamma-Ray Dosimeter

The "Phil" dosimeter was developed at ORNL to fill a need for a sensitive low-level gamma-ray dosimeter. A halogen-filled G-M counter is used as the detector. Hurst and Wagner developed the original version[I-1] as a neutron-insensitive dosimeter for use in mixed neutron and gamma-ray fields. The current version of this instrument utilizes an RCL 10-60 G-M tube with a 30 mg/cm^2-thick glass wall. Counters of this type have a "peaked" response to low-energy photons. An energy compensation filter consists of perforated layers of tin and lead. The inner layer is a 1-mm-thick sheet of tin containing 20 holes, each of which is 4-mm diam, and 24 holes of 4.7-mm diam. A 0.1-mm outer layer of lead contains 24 holes, each of which is 5-mm diam. The inherent background count rate for this detector is 40 counts/min. It is used in

the pulse mode in conjunction with a portable scaler. The response was found to be essentially constant (±12%) for photon energies down to 50 keV. Calibration of the instrument is performed using sealed NBS sources. A typical unit of this type yields 3400 counts/min for an exposure rate of 1 mR/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, an 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.

TECHNIQUES FOR THE MEASUREMENT OF RADON AND RADON DAUGHTERS IN AIR

Continuous Radon Monitor

Concentrations of radon are measured using a detector developed by Wrenn et al.[I-2] This detector operates on the principle that most of the 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 in the chamber 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′. Alpha pulses are counted and integrated for a fixed period of time, usually 30 min. At the end of each timed counting period, the total count for each channel is printed automatically and the system is reset and counting for the next period is initiated.

The radon monitor in use by ORNL is similar to that developed by Wrenn. However, the scintillation detector is larger (2 in. in diam), and a provision has been made to utilize an alpha source in order to standardize the chamber before putting it into service (see Fig. I-F). The alpha standard is inserted through a hole in the top of the chamber and rests in a fixed and repeatable position. During use of the monitor, the source access hole is plugged with a rubber stopper. An overall view of the ORNL radon monitor is shown in Fig. I-G.

Radon Progeny Monitor

An alpha spectrometry technique has been refined by Kerr[I-3,I-4] 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 concentrations 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 micron. A sampling time of 5 min and a flow rate of 12 liters/min are used. This filter sample is then placed under a silicon surface barrier detector and counter. The detector and counting system used for radon daughter measurements is shown in Fig. I-H. 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.[I-5] 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

$$\frac{dn_i(t)}{dt} = C_i v + g_{i-1}\, n_{i-1}(t) - g_i n_i(t), \tag{1}$$

where

n_i = number of the ith species of atom on the filter as a function of time,

g_i = radioactive decay constant of the ith species (min^{-1}),

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

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

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

$$y = e^{-ax}\,[y_0 + \int F(x)\, e^{ax} dx]. \tag{2}$$

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.

REFERENCES FOR APPENDIX I

I-1. G. S. Hurst and E. B. Wagner, "A Geiger-Mueller Gamma Ray Dosimeter with Low Neutron Sensitivity," *Health Phys.* 5, Nos. 1/2, pp. 20-26 April (1961).

I-2. M. E. Wren, H. Spitz, and N. Cohen, "Design of a Continuous Digital Output Environmental Radon Monitor," *IEEE Trans. Nucl. Sci.* 22, 645 (1975).

I-3. G. D. Kerr, *Measurement of Radon Progeny Concentrations in Air Alpha-Particle Spectrometry*, Oak Ridge National Laboratory Report ORNL/TM-4924 (July 1975).

I-4. G. D. Kerr, "Measurement of Radon Progeny Concentrations in Air," *Trans. Am. Nuc. Soc.* 17, 541 (1973).

I-5. P. T. Perdue, W. H. Shinpaugh, J. H. Thorngate, and J. A. Auxier, "A Convenient Counter for Measuring Alpha Activity of Smear and Air Samples," *Health Phys.* 26, 114 (1974).

ORNL-Photo 6705-76

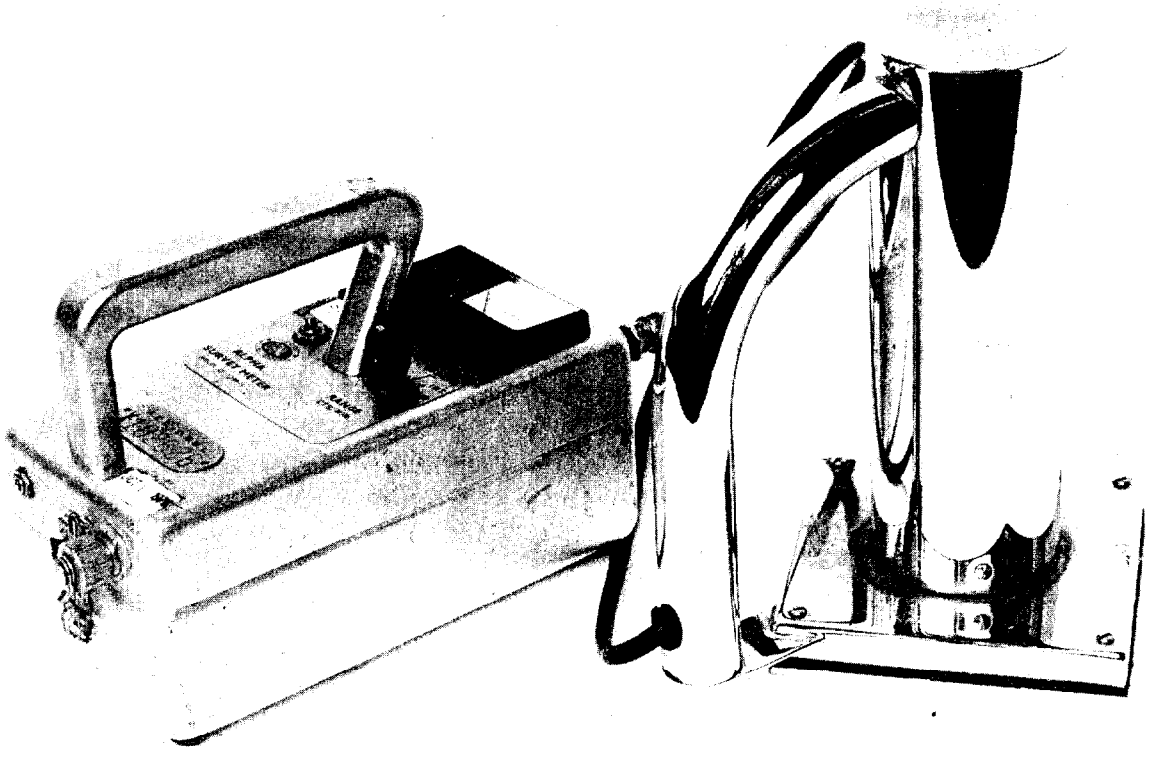

Fig. I-A. Alpha scintillation survey meter.

ORNL-Photo 6704-76

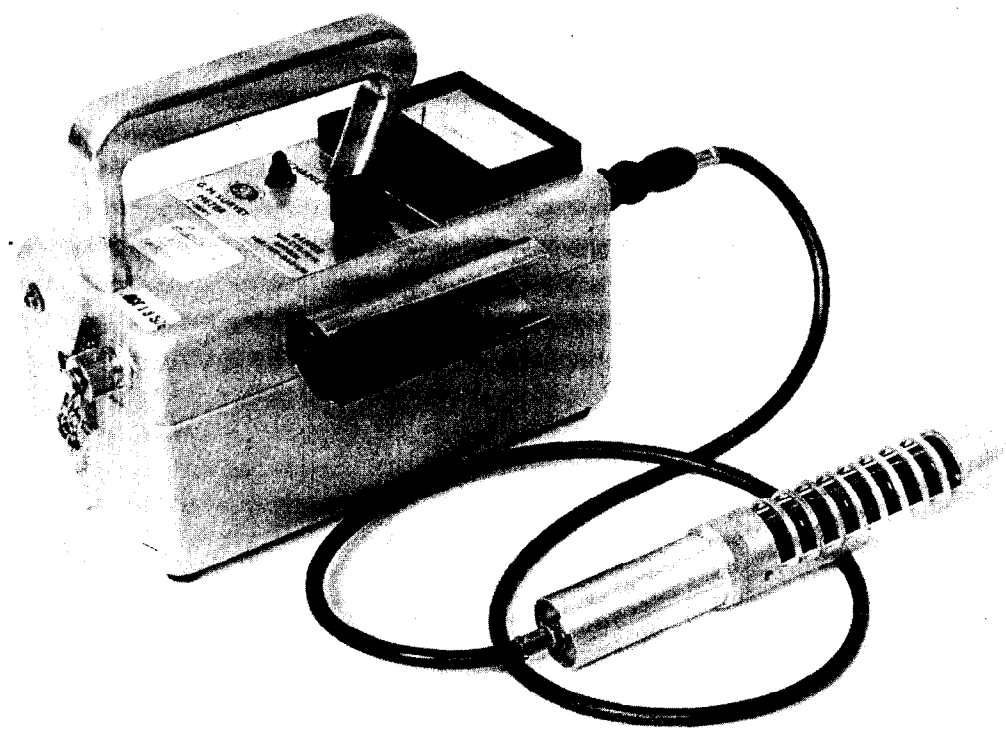

Fig. I-B. Geiger-Mueller survey meter.

ORNL-Photo 6710-76

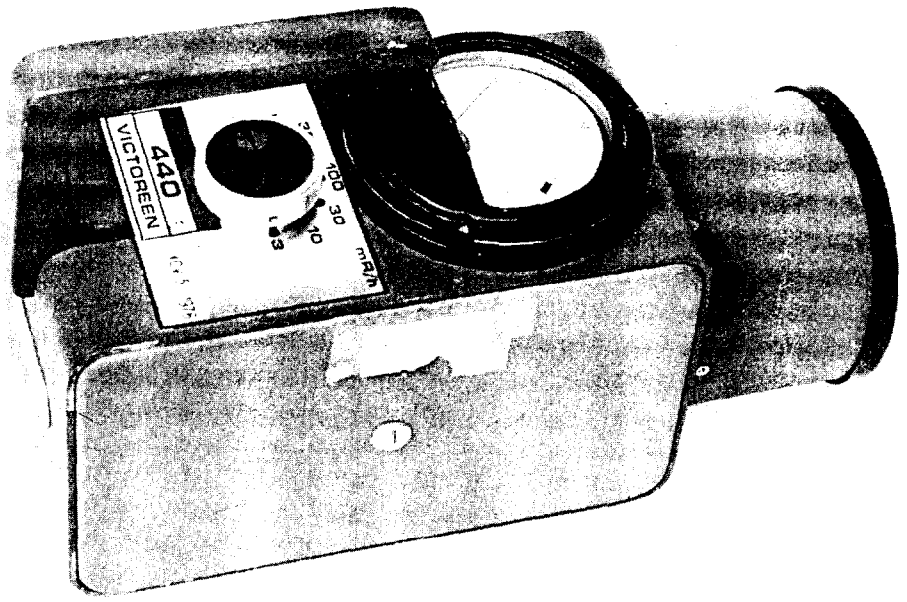

Fig. I-C. Victoreen Model 440 ionization chamber.

ORNL-Photo 6707-76

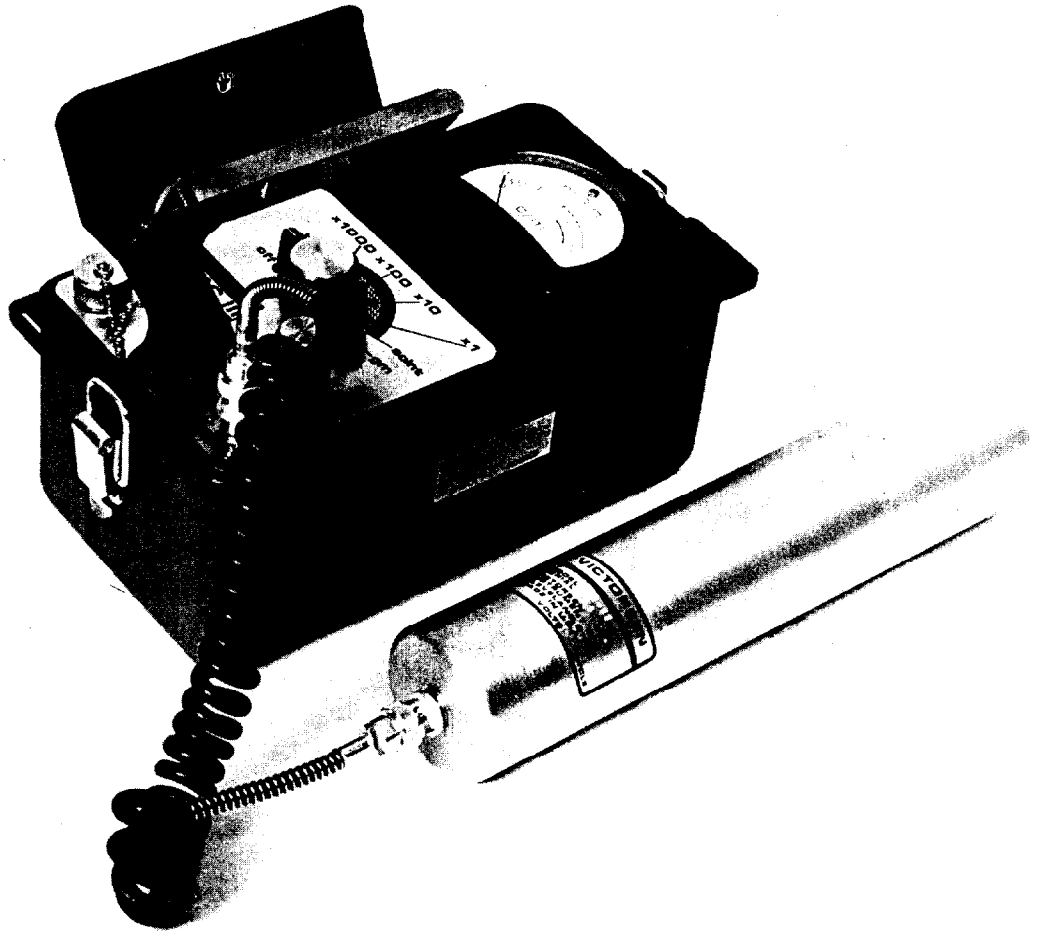

Fig. I-D. Gamma scintillation survey meter.

ORNL-Photo 1070-78

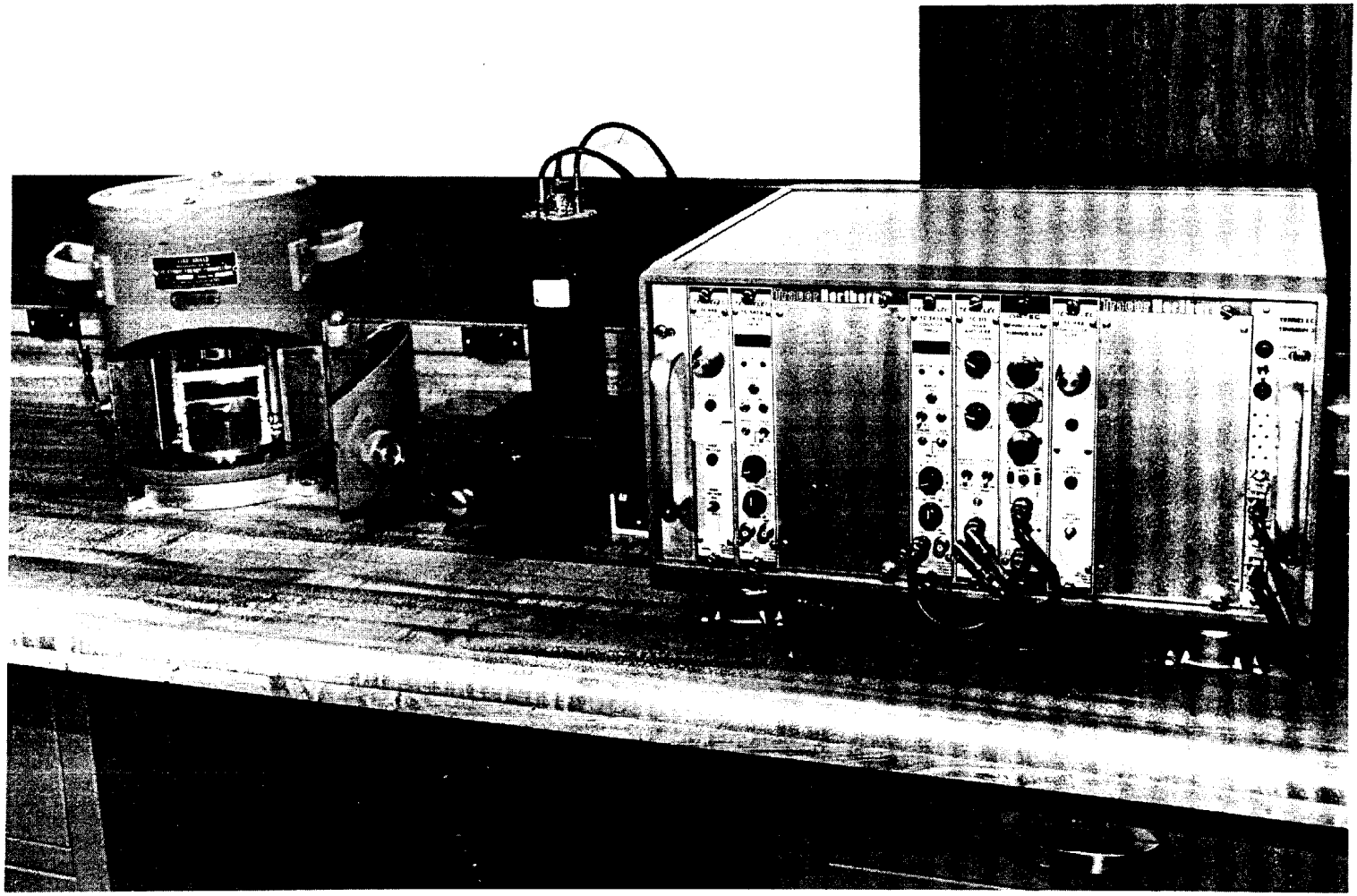

Fig. I-E. Smear counter and associated electronics. The beta counter is on the left and the alpha counter is on the right.

ORNL-Photo 0686-78

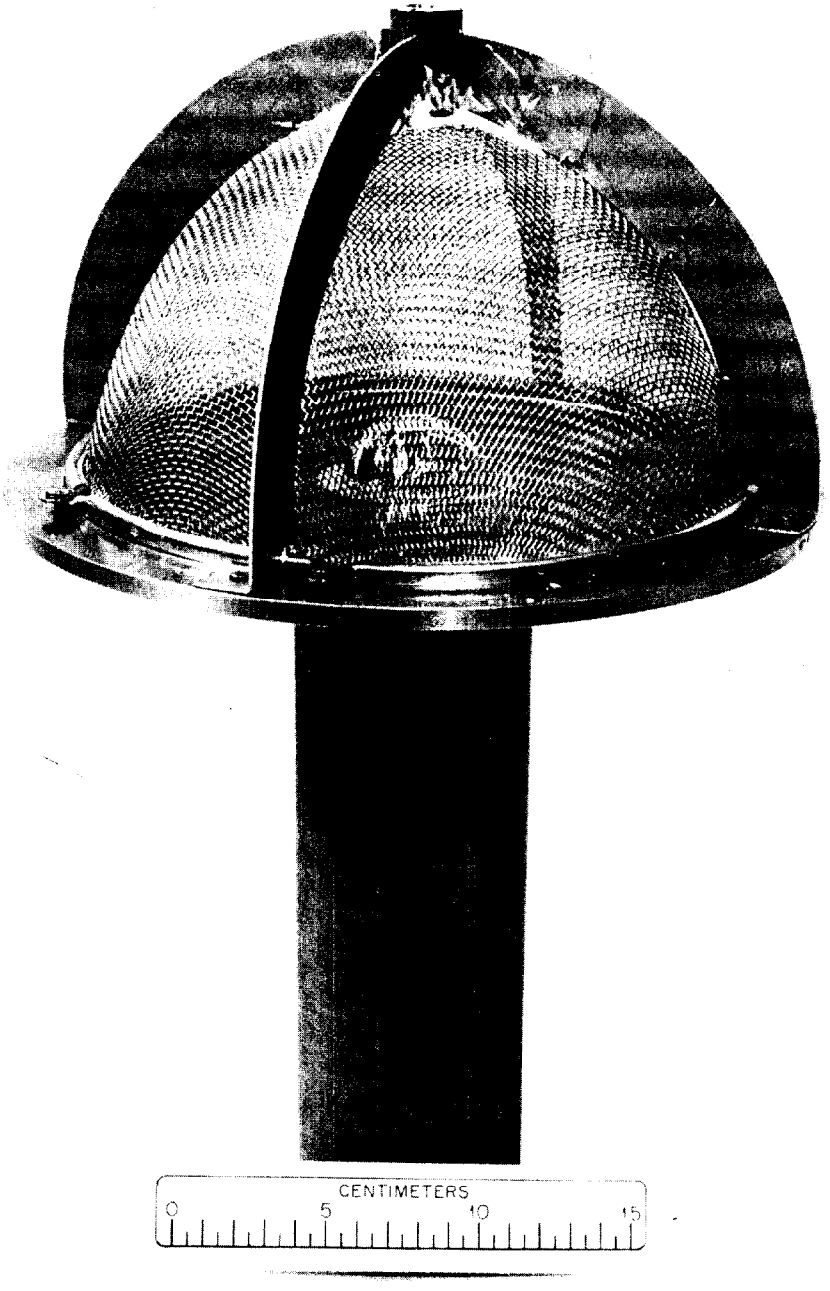

Fig. I-F. View of ionization chamber utilized in ORNL radon monitor. Shown is the photomultiplier housing, screen mesh hemisphere housing, and aluminized mylar covered ZnS scintillator.

ORNL-Photo 5199-78

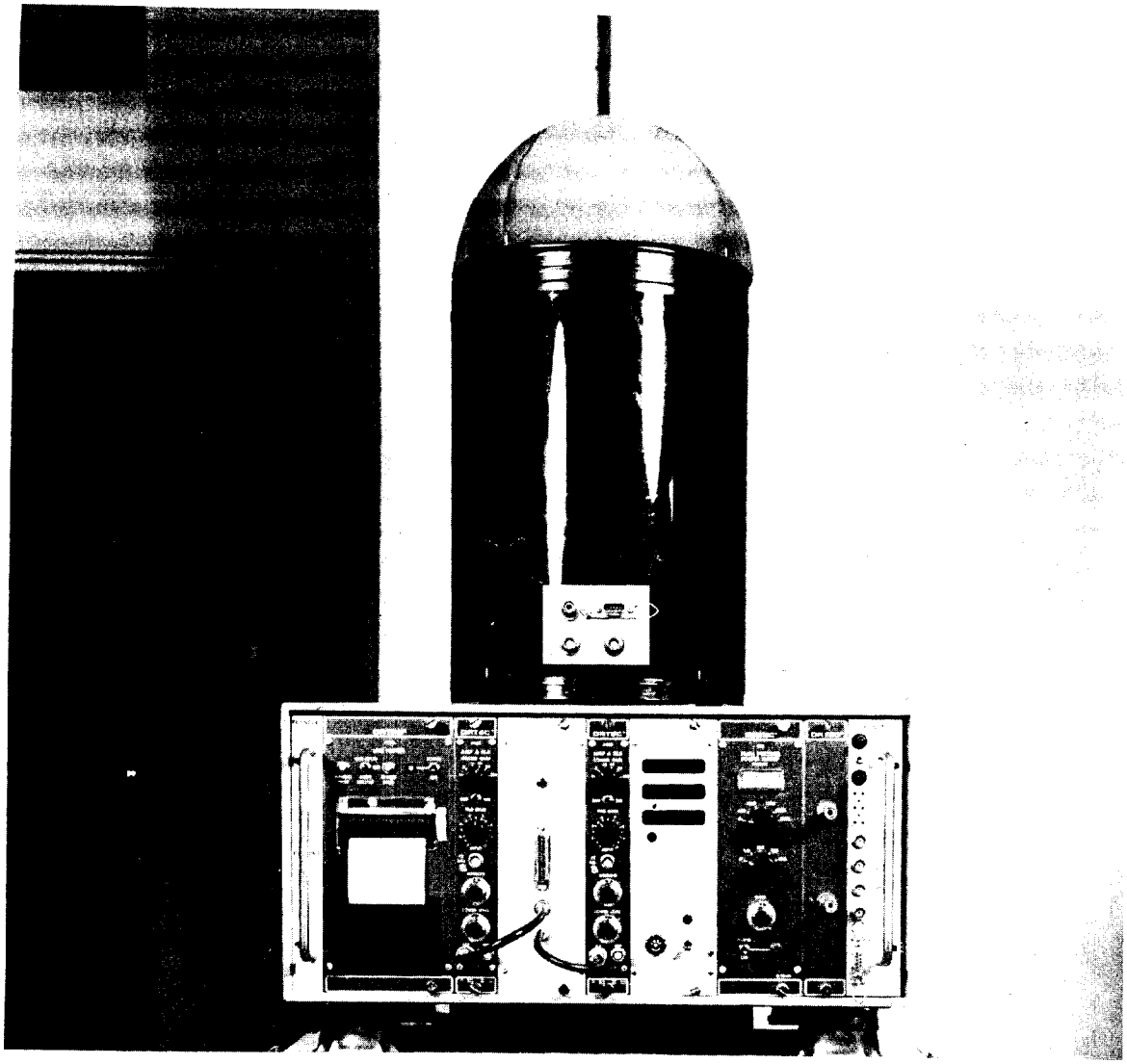

Fig. I-G. Overall view of ORNL continuous radon monitor.

ORNL-Photo 1077-78

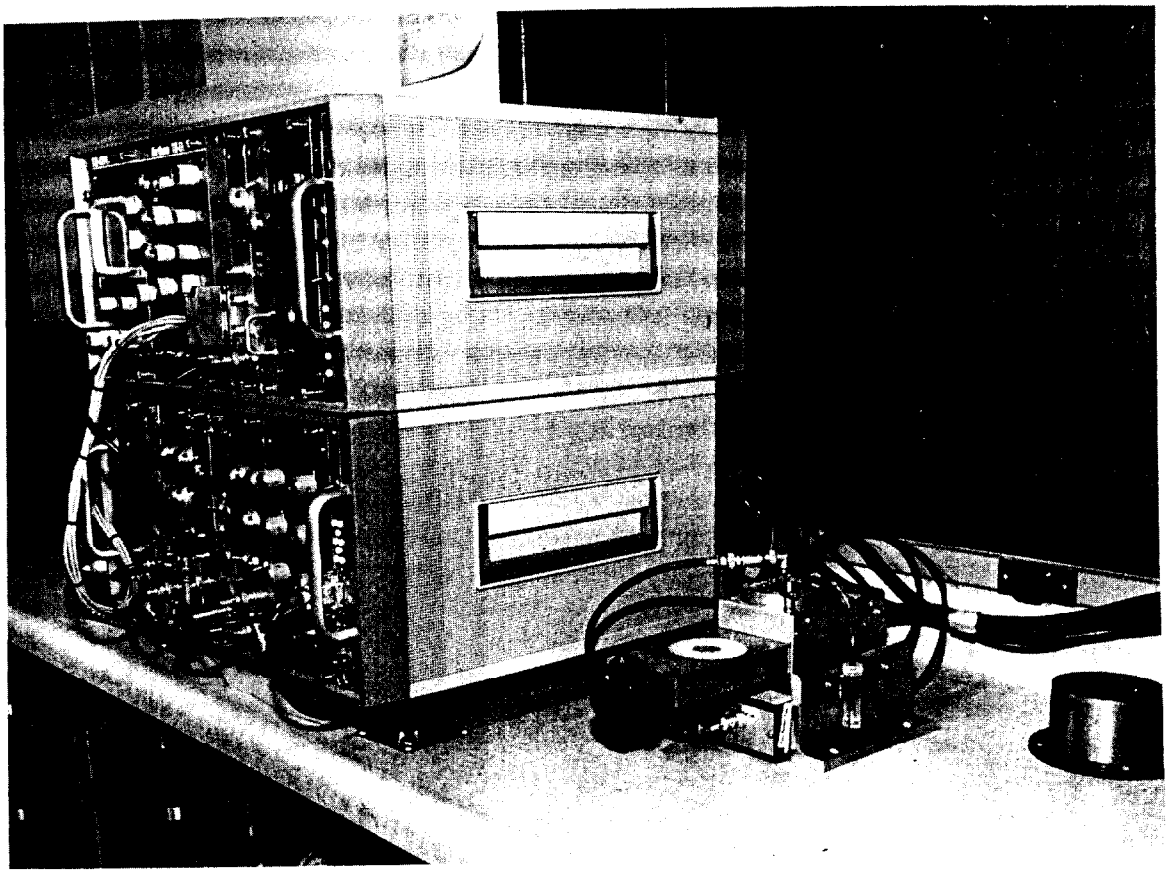

Fig. I-H. System used for measurement of radon daughter concentrations.

APPENDIX II

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 Fig. II-A). 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 300-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 computer based 4096-channel analyzer (see Fig. II-B), stored on magnetic tape, and subsequently analyzed by a computer program which uses an iterative least-squares method to identify radionuclides corresponding to those gamma-ray lines found in the sample. Also used is a 4096-channel analyzer with magnetic tape for mass storage and spectrum analysis using a program stored on the ORNL IBM computer system. 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.

ORNL Photo 2172-75

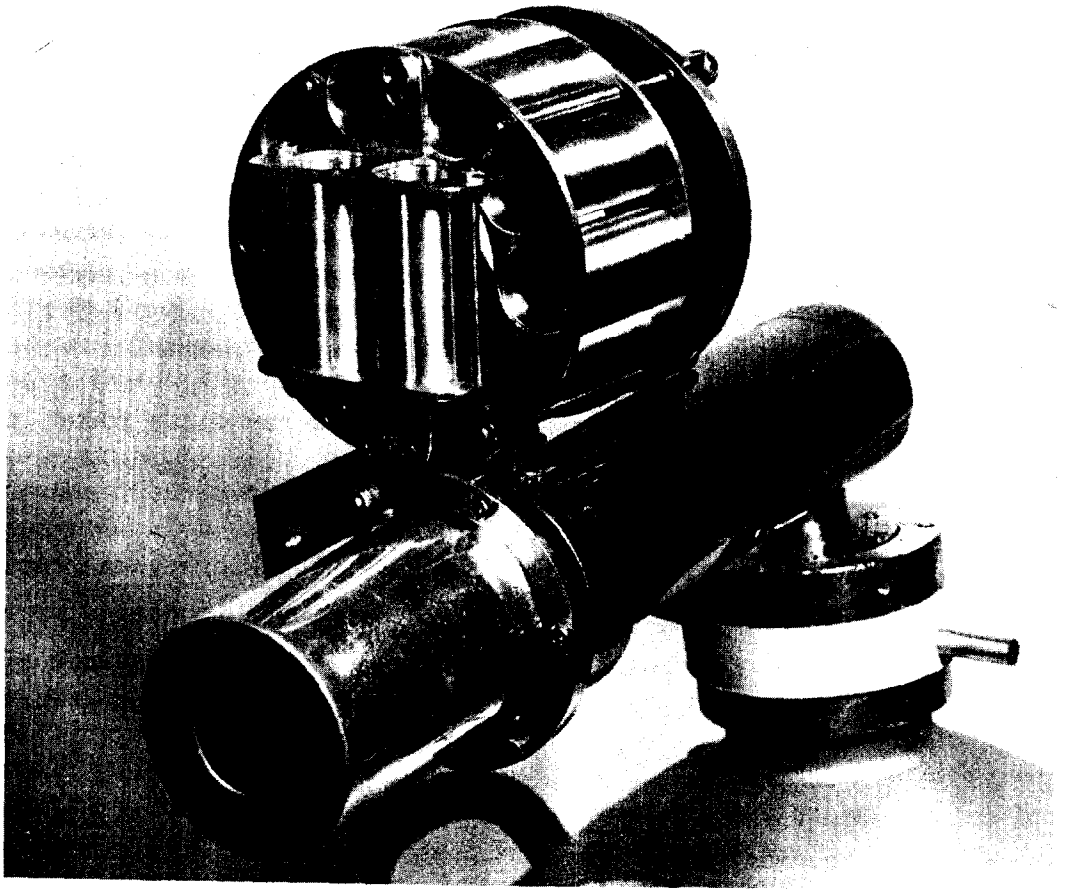

Fig. II-A. Holder for Ge(Li) detector system.

ORNL Photo 6719-76

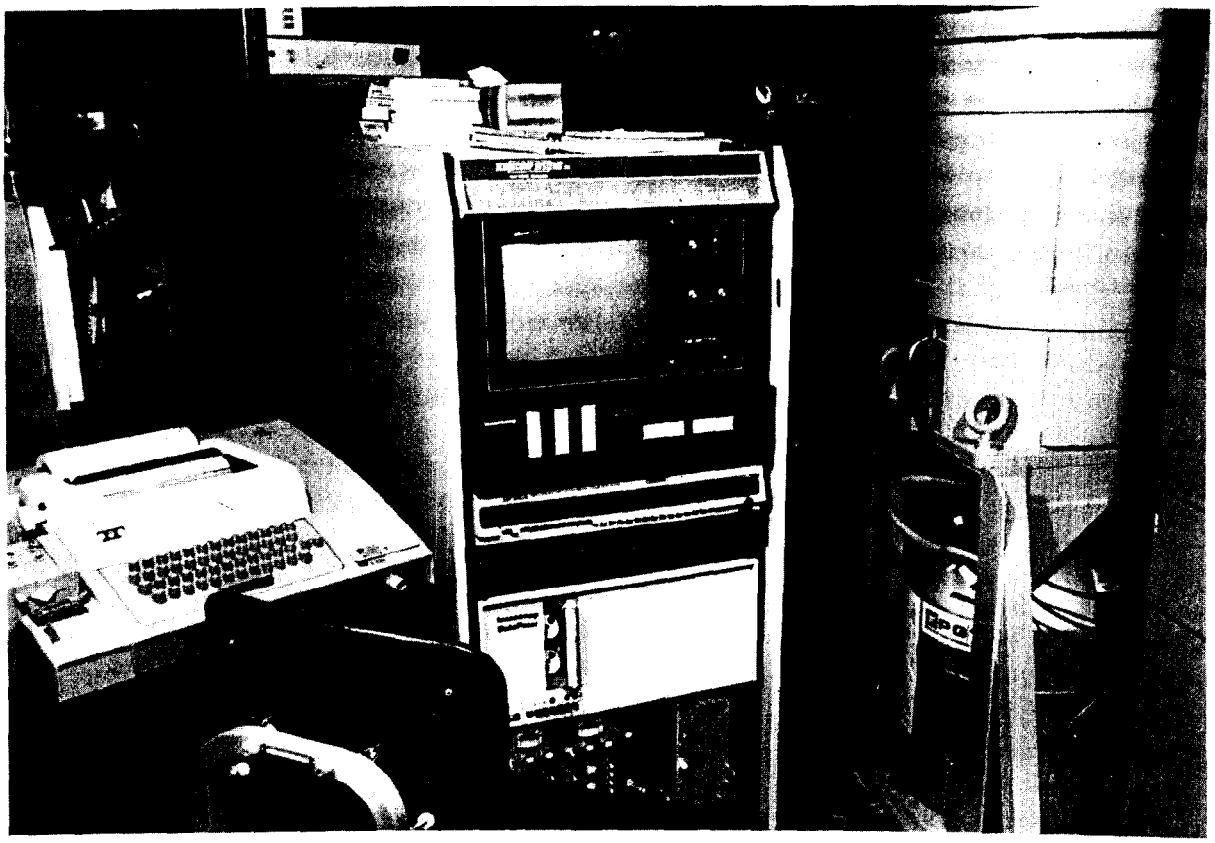

Fig. II-B. Computer-based 4096-channel analyzer.

APPENDIX III

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 III-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 III-1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control will be considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.
2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table III-1 prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.
3. The radioactivity on the interior surfaces of pipes, drain lines, or 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 material 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 material 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 III-1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also with 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 III-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 βγ/100 cm^2	15,000 dpm βγ/100 cm^2	1,000 dpm βγ/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.

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

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

[f] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/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.

Excerpts from
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 III-2 or Table III-3. (Table III-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 III-2. Surface contamination limits

The levels may be averaged[a] 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[b] is 2 x 10^{-13} Ci/m^3 or less or for which the nonoccupational MPC_w[c] is 2 x 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, -238.[d]	100	20
Group 2: Those nuclides not in Group 1 for which the nonoccupational MPC_a[b] is 1 x 10^{-12} Ci/m^3 or less or for which the nonoccupational MPC_w[c] is 1 x 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.[d]	1000	200
Group 3: Those nuclides not in Group 1 or Group 2.	5000	1000

[a] See note following table on applications of limits.

[b] 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 20, Appendix B, Table 2, Column 1).

[c] MPC_w: Maximum Permissible Concentration in Water applicable to members of the public.

[d] 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 occurrence of a 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 one.

Table III-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[a] 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.	1000	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.	5000	1000

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[a]Note on application of Tables III-2 and III-3 to isolated spots or activity:

For purposes of averaging, any m^2 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 $1/n \sum_{1}^{n} Si \geq L$, 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 $1/n \sum_{1}^{n} Si \geq AL$, 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 tailings which have been used as construction-related material.

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

712.2 Scope

The regulations in this part apply to all structures in the area of Grand Junction, 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 the Energy Research and Development Administration 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 Energy Research and Development Administration or 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-218), 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:

(1) For dwellings and schoolrooms:

(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.

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

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

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

712.10 Selection of appropriate remedial action

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

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

ENVIRONMENTAL PROTECTION AGENCY
Title 40, Part 141

Drinking Water Regulations--Radionuclides

Interim Primary Drinking Water Regulations
Promulgation of Regulations on Radionuclides
Federal Register, Vol. 41, No. 133, pp. 28402-9, Friday, July 9, 1976

Part 141.15 *Federal Register*
Vol. 41, No. 133, p. 28404, Friday, July 9, 1976

Maximum contamination levels for ^{226}Ra, ^{228}Ra, and gross alpha particle radioactivity.

(a) Combined ^{226}Ra and ^{228}Ra - 5 pCi/liter.

(b) Gross alpha particle activity (including ^{226}Ra but excluding radon and uranium) - 15 pCi/liter.

APPENDIX IV

EVALUATION OF RADIATION EXPOSURES AT THE FORMER VIRGINIA-CAROLINA CHEMICAL CORPORATION URANIUM RECOVERY PILOT PLANT, NICHOLS, FLORIDA

The U.S. Department of Energy (DOE) has determined that the former Virginia-Carolina Chemical Corporation Uranium Recovery Pilot Plant, Nichols, Florida, presently contains low levels of radioactive residues. The plant was designed to recover uranium from wet process phosphoric acid and to concentrate the uranium into a product containing at least 50 percent U_3O_8. The work was carried out under contract with the Atomic Energy Commission (AEC) from 1954 until 1960, at which time the pilot plant was disassembled. Tanks, piping, and other plant equipment used in the operation were removed; their present location could not be determined on the basis of available records. The only remaining structure consists of a concrete pad which is currently situated within the boundaries of a phosphate products plant operated by Conserv, Inc. A small building has been constructed on an adjacent concrete pad and consists of a maintenance shop, lunchroom, tool storage cage, and a small office.

A presurvey inspection of the site was conducted in April, 1977 and indicated elevated levels of beta-gamma radiation and alpha activity in the vicinity of the concrete pad. Soil samples taken adjacent to the pad showed elevated levels of radium-226 and uranium-238. Shortly after the initial visit, a layer of soil totaling approximately three cubic meters was removed by the plant operator and transferred to an inactive gypsum pile approximately 800 meters from the plant. The material was covered with an estimated 0.7 to 1.0 meter of gypsum and soil.

The contamination from previously contracted operations is producing slight radiation exposures to employees working at this site. These exposures result primarily from gamma and beta radiation emitted by contamination in the ground or on the surface of the concrete pad. The additional exposures received by ingestion (e.g., eating or drinking in one of the buildings) or by the inhalation of airborne activity are relatively small as compared with exposures to direct gamma and beta radiation. A summary of radiation exposures at the former Virginia-Carolina site is provided in Table IV-1 along with appropriate guidelines and background values.

The naturally occurring radionuclides present at the site are also present in minute quantities throughout our envirnoment. Concentrations

of these radionuclides in normal soil, air, water, food, etc., are referred to as background concentrations. Radiation exposures resulting from this environmental radioactivity are referred to as background exposures. These background exposures are not caused by any human activity and, to a large extent, can be controlled only through man's moving to areas with lower background exposures. Each and every human receives some background exposure daily.

The use of radioactive materials for scientific, industrial, or medical purposes may cause radiation exposures above the background level to be received by workers in the industry and, to a lesser extent, by members of the general public. Scientifically based guidelines have been developed to place an upper limit on these additional exposures. Limits established for exposures to the general public are much lower than the limits established for workers in the nuclear industry.

Uranium-238 is believed to have been created when the earth was formed. It is still present today because it takes a very long time to decay. The half-life is a measure of the time required for radioactive decay; for uranium-238 it is 4.5 billion years. Thus, if you begin with one curie* of uranium-238, one-half curie will remain after 4.5 billion years. After 9 billion years, this would only be one-fourth curie of uranium-238, etc. As the uranium-238 decays, it changes into another substance--thorium-234. Thorium-234 is called the "daughter" of uranium-238. In turn, thorium-234 is the "parent" of protactinium-234. Radioactive decay started by uranium-238 continues as shown in Table IV-2 until stable lead is formed. The "decay product" listed in Table IV-2 is the radiation produced as the parent decays.

Direct Gamma and Beta-Gamma Exposure

As may be seen in Table IV-2, several of the daughters of uranium-238 and of radium-226 emit gamma radiation. (Gamma rays are penetrating radiation like X-rays.) Hence, the contaminated areas can be sources of external gamma radiation exposure.

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*A curie is a unit defined for expressing the amount of radioactivity present in a substance; one curie represents 37 billion radioactive disintegrations per second.

Evaluations of the gamma radiation exposures resulting from the presence of the former uranium recovery operations are difficult to assess due to a layer of gypsum dust which has deposited on the site. This gypsum dust results from current phosphate-rock operations at the plant. The contribution of this dust to any measurement of radioactivity in the surveyed area must be taken into account because of the elevated radium-226 content normally present in this dust. Several background external gamma readings were taken in the Conserv plant at locations sufficiently removed from the survey area to be unaffected by the pilot plant operations. These measurements indicate an average external gamma radiation level (on site) of 41 microRoentgens per hour.* Measurements in the survey area ranged from 20 to 100 microRoentgens per hour and averaged 55 microRoentgens per hour. External gamma radiation levels in the existing building adjacent to the concrete pad averaged 12 micro-Roentgens per hour in the maintenance room and tool cage, and 18 micro-Roentgens per hour in the lunchroom and office. Thus, exposure to the average external gamma radiation level in the lunchroom, 18 microRoentgens per hour, for 2,000 hours per year (a normal work year) would produce an annual exposure of 36,000 microRoentgens. For comparison, a typical chest X-ray (according to Department of Health, Education, and Welfare data) might yield an exposure of about 27,000 microRoentgens.

The National Council on Radiation Protection and Measurements has recommended a maximum annual whole-body exposure of 500,000 microRoentgens per year to an individual continually exposed in the general population; this would correspond to exposure to 250 microRoentgens per hour for 2,000 exposure hours. External gamma radiation exposures at this site are well below this guideline value. The guideline for exposure to an individual in the general public is ten times lower than guidelines established for a worker in the nuclear industry.

The surface of the concrete pad is contaminated with both uranium-238 and radium-226. Nuclear Regulatory Commission (NRC)

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*The Roentgen is a unit which was defined for radiation protection purposes for people exposed to penetrating gamma radiation. A micro-Roentgen is one-millionth of a Roentgen.

guidelines state that the average beta-gamma dose rate measured at a distance of one centimeter above surfaces should not exceed 0.2 millirad* per hour. Contamination on the pad results in beta-gamma dose rates in excess of 0.2 millirad per hour at two locations, with a maximum level of 0.26 millirad per hour. In addition, beta-gamma dose rates at two locations in the area surrounding the pad exceed 0.2 millirad per hour with a maximum value of 0.75 millirad per hour near the northwest corner of the pad. Thus, handling this material for a period of one hour could produce a skin dose of 0.75 millirad. For comparison, the skin dose which would be expected from a normal year's watching of color television by an adult is 1.6 millirads; for a child less than 15 years of age, the comparable dose is 3.6 millirad per year (according to the United Nations Scientific Committee on the Effects of Atomic Radiation).

The primary concern of the NRC guideline is exposure of skin surfaces. The thickness of ordinary shoe soles is adequate to protect the skin of feet from beta radiation. Other areas of body skin are adequately protected from these exposures if they remain away from these surfaces. In most cases, exposures are negligible at a distance of one foot away from these surfaces. Although potential exists for exposures in excess of the guidelines, beta and gamma surface exposures are believed to be inconsequential to employees at this site due principally to a low frequency of occupancy.

Exposure from Inhalation of Radionuclides in Air

The deposits of radium-bearing residues in soil are the indirect sources of some of the radiation exposure of persons on the site. As may be seen in Table IV-2, radium-226 changes to radon-222 as a result of radioactive decay. Radon-222 is an inert gas which can leave the ground and seep through floors into buildings. If not diluted by additional ventilation air, the concentration can build up in closed areas of the buildings. As indicated in Table IV-2, workers on this site are exposed to average radon concentrations which are typical for the

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*The millirad is a unit used to measure the amount of radiation energy absorbed in human tissue.

background level in the surrounding areas. These concentrations are well below the recommended guideline for exposure of the general public given in Federal Regulation 10 CFR 20.*

Radioactive decay of radon-222 is rapid (days), and its decay gives rise to short-lived daughters as shown in Table IV-2. Background concentrations of radon daughters both inside and outside structures are typically less than 0.01 working level (WL).† Exposures to radon daughters at this site cannot be distinguished from background.

Studies of uranium and other hard-rock miners have established that inhalation of large quantities of daughters of radon-222 over long periods of time increases an individual's risk of contracting lung cancer. The present federal guide value for uranium mine workers (given by the Environmental Protection Agency [EPA]), when translated to the units discussed here, would limit mine workers to an exposure of 0.33 working levels, assuming exposure for 2,000 hours per year, a typical work year. This level is significantly lower than the exposures received by most of the miners included in the studies mentioned above.

Other Considerations of Exposure

The concentration of radionuclides in all three water samples taken on site were below the concentration guide for water (CG_w) set forth in 10 CFR 20.

While no crops are currently grown on this site, use of the contaminated soil for such purposes could produce additional human exposure through consumption of crops which have incorporated radium-226 or other radionuclides. In addition, actions which involve considerable scraping or tilling of dry soil, particularly in the areas showing high concentrations of radium-226 in surface soil, could lead to human exposures through inhalation of airborne radioactive dust.

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*Title 10, Code of Federal Regulations, Part 20, is a document published by the Nuclear Regulatory Commission and may be found in the *Federal Register*.

†The working level is a unit which was defined for radiation protection purposes for uranium miners. It represents a specific level of energy emitted by the short-lived daughters of radon.

If buildings were to be constructed over areas having the highest concentrations of radium-226 in soil, radon concentrations higher than those measured on site could exist in those buildings. Furthermore, the guidelines given in 10 CFR 712* for radon daughters could be exceeded in these buildings. Consequently, careful consideration should be given to the location of any structure built on this site in the future.

Risk and Radiation Exposures

Risks resulting from radiation exposures should be considered within the context of other risks incurred in normal living. For simplicity, risks to health may be classified in four categories:

1. Unacceptable--problems with risk so high as to require immediate action, such as severe diseases where medical treatment is required to save a life.
2. Concerned--problems where people are willing to spend time and money to reduce potential hazards. Examples of this include the maintenance of public highways and signs, signals, fire departments, and rescue squads.
3. Recognized--problems where people may accept some inconvenience to avoid certain activities such as flying in airplanes, swimming alone, etc.
4. No great concern--problems with a low frequency of occurrence. There is an awareness of potential hazard, but an accompanying feeling that these problems occur only to other people.

An individual may be exposed to risks over which he can exercise some control (voluntary), and risks over which he feels he has no personal control or choice (involuntary).

Daily, an individual is confronted with decisions about risk which have an associated benefit--for example, driving a car. This can serve as an illustration that a voluntary, concerned risk may be deemed appropriate due to the desirable perceived benefit. As another example,

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*Title 10, Code of Federal Regulations, Part 712, is a document published by the Energy Research and Development Administration (now Department of Energy) and may be found in the *Federal Register*.

an individual who smokes cigarettes has subjected himself to a risk of lung cancer which is about ten times higher than that for a nonsmoker.

For purposes of radiation protection, all radiation exposures are assumed to be capable of increasing an individual's risk of contracting cancer. A precise numerical value cannot be assigned with any certainty to a given individual's increase in risk attributable to radiation exposure. The reasons for this are numerous; they include the individual's age at onset of exposure, variability in latency period (time between exposure and physical evidence of disease), the individual's personal habits and state of health, previous or concurrent exposure to other cancer-causing agents, and the individual's family medical history. Because of these variables, large uncertainties would exist in any estimates of the number of increased cancers in the relatively small working population at the Conserv site.

The normal annual death rate* from lung cancer for all population groups in Polk County (as of 1970) was 22.7 deaths per 100,000 population. At the same time, the annual death rate from lung cancer for all population groups in the United States and the state of Florida were 21.1 and 23.0 deaths per 100,000 population, respectively. A one-year exposure to the guideline value for uranium miners (0.33 working level for 2,000 hours) might increase the risk of death due to lung cancer by approximately four percent.

The annual death rate from all types of cancer among all population groups in Polk County (as of 1970) was 133 deaths per 100,000 population. At the same time, the death rate from all types of cancer for all population groups in the United States and in the state of Florida were 151 and 150 per 100,000 population, respectively. A one-year exposure to penetrating gamma radiation of 500,000 microRoentgens might increase the risk of death due to all types of cancer by about one-tenth of a percent. Exposures in excess of these guideline values would be expected to result in proportionately higher increases in risk. Consequently,

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*Mortality statistics were obtained from data in *U.S. Cancer Mortality by County: 1950-1969*, prepared by the National Cancer Institute, 1973, available from the U.S. Government Printing Office.

any action taken to reduce either the rate or the duration of radiation exposures would also reduce the risk attendant to that exposure.

Remedial Measures

That fraction of the total radiation exposures which results from the former usage of the Conserv site as a uranium recovery pilot plant arises from contamination of the concrete pad and surrounding soil by radium-226 and uranium-238. This contamination is mainly in the top 1-1/2 feet of soil, with the result that removal of this layer followed by replacement with clean overfill would reduce present and potential radiation exposures associated with this contamination. In addition, cleaning and/or removal of the concrete pad would serve to lower further the small exposures resulting from surface contamination on the pad.

SUMMARY

The former Virginia-Carolina Chemical Corporation Uranium Recovery Pilot Plant is contaminated with residues containing low-level concentrations of uranium-238 and radium-226. This contamination is primarily contained in the top 1-1/2 feet of soil in areas surrounding the concrete pad and in surface deposits on the pad itself. External gamma exposure rates have been slightly elevated due to this contamination, as have the beta-gamma dose rates at the pad surface. However, the contamination has the potential to produce radiation exposures which could approach, and in some cases possibly exceed, scientifically based guidelines. Consequently, some remedial measures are in order. The Department of Energy has developed a coordinated plan which addresses specific problems at facilities such as the former Virginia-Carolina plant. Currently, work is underway to implement the elements of this plan.

Table IV-1. Summary of exposure data at the former Virginia-Carolina Chemical Corporation Uranium Recovery Pilot Plant, Nichols, Florida

Exposure source	Background levels	Guideline value for general public	Guideline value for radiation workers	Average levels at Nichols site
Radon in air	Less than 1 picocurie[a] per liter of air	Continuous exposure to 3 picocuries per liter of air	Exposure for 40 hours per week and 50 weeks per year to 30 picocuries per liter of air	Average daytime concentration ranged from 0.6 to 0.9 picocurie per liter of air
Radon daughters in air	Less than 0.01 working level[b]	0.01 working level for residences and school rooms, and 0.03 working level for other structures	0.33 working level for uranium miners exposed for 40 hours per week and 50 weeks per year	Less than 0.01 working level
Gamma radiation from daughters of radium and uranium contamination	11 micro-Roentgens[c] per hour in the Nichols area	250 microRoentgens per hour above natural background for 40 hours per week and 50 weeks per year for an individual. This is equivalent to 0.5 Roentgen per year	2500 microRoentgens per hour for 40 hours per week and 50 weeks per year. This is equivalent to 5 Roentgens per year	Average gamma radiation levels 1 meter above the floor or ground ranged from 12 to 55 microRoentgens per hour

[a]The picocurie is a unit which was defined for expressing the amount of radioactivity present in a substance.

[b]The working level is a unit which was defined for radiation protection purposes for uranium miners. It represents a specific level of energy emitted by the short-lived daughters of radon.

[c]The Roentgen is a unit which was defined for radiation protection purposes for people exposed to penetrating gamma radiation. A microRoentgen is one-millionth of a Roentgen.

Table IV-2. Uranium-238 decay series

Parent	Half-life	Decay products	Daughter
uranium-238	4.5 billion years	alpha	thorium-234
thorium-234	24 days	beta, gamma	protactinium-234
protactinium-234	1.2 minutes	beta, gamma	uranium-234
uranium-234	250 thousand years	alpha	thorium-230
thorium-230	80 thousand years	alpha	radium-226
radium-226	1600 years	alpha	radon-222
radon-222	3.8 days	alpha	polonium-218
polonium-218[a]	3 minutes	alpha	lead-214
lead-214[a]	27 minutes	beta, gamma	bismuth-214
bismuth-214[a]	20 minutes	beta, gamma	polonium-214
polonium-214[a]	$\frac{2}{10,000}$ second	alpha	lead-210
lead-210	22 years	beta	bismuth-210
bismuth-210	5 days	beta	polonium-210
polonium-210	140 days	alpha	lead-206
lead-206	stable	none	none

[a]Short-lived radon daughters.

*U.S. GOVERNMENT PRINTING OFFICE:1980 310-912/191 1-3