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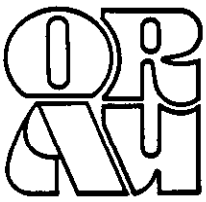

Prepared by
Oak Ridge Associated
Universities

Prepared for
U.S. Nuclear
Regulatory
Commission's
Region I Office

Supported by
Safeguards and
Materials Program
Branch;
Division of
Quality Assurance,
Safeguards, and
Inspection Programs;
Office of
Inspection and
Enforcement

# CONFIRMATORY RADIOLOGICAL SURVEY

# OF THE

# URANIUM OXIDE LABORATORY

# WESTINGHOUSE NUCLEAR FUEL DIVISION

# CHESWICK, PENNSYLVANIA

**M. W. STAFFORD**

Radiological Site Assessment Program
Manpower Education, Research, and Training Division

FINAL REPORT

May 1984

# CONFIRMATORY RADIOLOGICAL SURVEY OF THE URANIUM OXIDE LABORATORY WESTINGHOUSE NUCLEAR FUEL DIVISION CHESWICK, PENNSYLVANIA

Prepared for

Safeguards and Materials Program Branch
Division of Quality Assurance, Safeguards, and Inspection Programs
U.S. Nuclear Regulatory Commission
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FINAL REPORT

May 1984

This report is based on work performed under Interagency Agreement DOE No. 40-816-83, NRC Fin. A-9076-3, between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy. Oak Ridge Associated Universities performs complementary work under contract number DE-AC05-76OR00033 with the U.S. Department of Energy.

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# CONFIRMATORY RADIOLOGICAL SURVEY OF THE URANIUM OXIDE LABORATORY WESTINGHOUSE NUCLEAR FUEL DIVISION CHESWICK, PENNSYLVANIA

## INTRODUCTION

From 1959 through 1979, the Nuclear Fuel Division of the Westinghouse Electric Corporation performed process development and pilot fabrication of reactor fuels at the Cheswick, Pennsylvania, site. Operations in the Uranium Oxide Laboratory, Building 7, were terminated in 1979 and decontamination and decommissioning efforts were initiated. Upon completion of the final radiological survey of the facility, the licensee presented results in an August 1983 report submitted to the Nuclear Regulatory Commission (NRC). The licensee's survey findings indicated the radiological conditions satisfied the criteria for release for unrestricted use. At the request of the Nuclear Regulatory Commission, the Radiological Site Assessment Program of Oak Ridge Associated Universities (ORAU) conducted a confirmatory survey of the previous Uranium Oxide Laboratory site during September 1983. This report presents the procedures and results of that survey.

## SITE DESCRIPTION

The Cheswick Operations facility of Westinghouse Electric Corporation is located on Low Grade Road, approximately 0.5 km northwest of Cheswick, Pennsylvania (see Figure 1). Buildings on this site, in which radioactive materials were handled, included: Building 6, the Astronuclear Laboratory; Building 7, housing the Uranium Oxide Laboratory and the Plutonium Development Laboratory; Building 8, the Plutonium Fuels Development Laboratory; and Building 9, the Naval Pump Repair Facility. Building 6 was previously decommissioned and has been demolished; the Plutonium Development Laboratory area of Building 7 has also been decommissioned and Building 8 was in the process of being decontaminated at the time of this survey. The Naval Pump Repair Facility adjacent to Building 7 continues in operation. Building 10 served as a security and receiving facility of Building 8 and has more recently been used for storage and to house a laboratory in support of decommissioning activities.

The Uranium Oxide Laboratory, located in the west end of Building 7, (see Figure 2) processed uranium oxide powder, ranging from depleted to fully enriched. Small quantities of thorium oxide were also used. Material in powder form was processed in hoods or closed containers; pressed or sintered pellets were handled in the open laboratory atmosphere.

The facility (see Figure 3), occupying approximately 420 $m^2$, is constructed with concrete block walls, a reinforced concrete floor, and a sheet-steel roof covered with tarred felt. The majority of the powdered fuel handling operations were performed in a high-bay area (Room C); a smaller high-bay area (Room D) was used for pressing and sintering fuel pellets. Other rooms were primarily for support activities. Drain lines, throughout the facility, carried contaminated liquid to the waste storage pit on the north side of the building.

Equipment, ventilation systems, and interior non-load bearing walls have been removed. The entire floor drain system has been removed and open trenches remain. Figure 4 is the floor plan of the building showing remaining interior walls and the locations of excavated floor areas.

## SURVEY PROCEDURES

During the period of September 12-16, 1983, ORAU personnel conducted a confirmatory radiological survey of the uranium laboratory. The purpose of this survey was to verify the adequacy and accuracy of the licensee's final survey and confirm the radiological condition of the site relative to decommissioning criteria.

### Objectives

The objectives of this survey were to:

1. Measure direct radiation levels in the facility,

2. Measure fixed and removable surface contamination of floors, ceiling areas, walls, miscellaneous fixtures, sumps, drains, etc., and

3. Determine radionuclide concentrations in soil and residues from excavated areas and trenches.

## Procedures

### Gridding

The grid system established by the licensee, i.e. 1 m x 1 m, on floors and walls and approximately 0.6 m x 1.5 m (2 ft x 5 ft) on the ceiling were used to reference measurements whenever possible. In areas where decontamination activities had removed the original grid system, ORAU reestablished the grid. Figure 5 shows this grid system in areas where residual contamination was identified.

### Surface Scanning

All grid blocks on the floor and lower walls (to 2 m) were surface scanned using low-energy gamma scintillation and pancake G-M detectors. Overhead surfaces were scanned at approximately 1 m intervals. Particular attention was given to scanning cracks, joints, and horizontal overhead surfaces such as ledges, beams, and pipes. Locations of elevated direct radiation levels were noted.

### Exposure Rate Measurements

Gamma exposure rates at 1 m above the floor were measured throughout the facility using a pressurized ionization chamber.

### Surface Contamination Measurements

Grid blocks where the licensee's report indicated contamination levels exceeding 25% of the guideline values, were selected for surveys for total and transferable alpha and beta-gamma contamination. Measurements were also performed on 100% of the floor and wall grid blocks in the waste storage pit, in areas where surface scans identified elevated levels, and on approximately 10% of the remaining grid blocks.

In each grid block surveyed, direct measurements of alpha and beta-gamma contamination levels were systematically performed at the center and four equidistant points, midway between the center and block corners. The five measurements were averaged and compared to the release criteria. Smears for transferable alpha and beta contamination were performed at the location in each grid block, where the highest direct measurement was obtained. Grid blocks, selected for measurements, were scanned and total and transferable contamination were determined at locations of elevated levels. Direct and transferable contamination measurements were performed on ledges, piping, and ungridded horizontal and vertical surfaces.

Soil Sampling in Excavated Areas

Trenches and open pits were scanned with a low-energy gamma scintillation detector and locations of elevated readings were noted. Soil samples were collected at elevated locations disclosed by scanning and at additional locations, to provide representative coverage of the excavated areas. Locations of soil sampling are shown on Figures 5 and 7.

Additional Sampling

Residue was collected from the sump of the storage pit and from a buried pipe, which indicated an elevated direct radiation level.

Comparison of Instrumentation used by Westinghouse and ORAU

Measurements of direct alpha and beta-gamma levels were performed by Westinghouse and ORAU personnel at several locations. Results were compared to evaluate the differences in instrument response.

Sample Analysis and Interpretation of Data

Smears were counted to determine gross alpha and gross beta activity. Soil and pipe residue samples were analyzed by gamma spectrometry for U-238, U-235, and any other identifiable photopeaks. Major analytical equipment used

for this survey is listed in Appendix A. Appendix B contains a description of the analytical procedures applicable to this survey.

Results were compared with guidelines for release of facilities for unrestricted use. Surface contamination guidelines established by the Nuclear Regulatory Commission are presented in Appendix C. Because natural thorium and enriched uranium isotopes were used in the facility, a combination of the thorium and uranium guidelines was specified by the NRC for application in this facility. These contamination guidelines are:

1. Alpha

   Total - 1000 dpm/100 $cm^2$, average*; 3000 dpm/100 $cm^2$ maximum
   Transferable - 200 dpm/100 $cm^2$

2. Beta-Gamma

   Total - 5000 dpm/100 $cm^2$, average*; 15,000 dpm/100 $cm^2$ maximum
   Transferable - 1000 dpm/100 $cm^2$

3. Uranium in Soil - 30 pCi/g

## RESULTS

### Exposure Rates

Exposure rate measurements throughout the building ranged from 9.5 to 13 μR/h. No significantly elevated levels were detected. Exposure rates were approximately 10% greater in excavated areas.

### Surface Contamination Levels

The results of surface contamination measurements are summarized in Table 1. Small areas of beta-gamma contamination, exceeding the maximum were

*Averaged over an area of 1 $m^2$.

identified on the floor of rooms A, C, D, and E. The highest level was 161,000 dpm/100 $cm^2$ in room A. Several of these locations also had total alpha levels above 3000 dpm/100 $cm^2$. One small area of the ceiling of room D was contaminated to 45,800 beta-gamma dpm/100 $cm^2$. This area was at the location of a ceiling opening, through which ventilation ductwork had passed.

Larger areas of surface contamination were identified in the waste storage pit. About 30% of the floor, 60% of the south wall, remaining lower walls, and the metal cover plates were all contaminated above the beta-gamma limits. Highest levels were in the vicinity of the sump and on residues remaining in the sump pit. No areas of transferable contamination above the alpha and beta-gamma limits of 200 dpm/100 $cm^2$ and 1000 dpm/100 $cm^2$, respectively, were noted.

The contaminated areas were identified and decontamination of several locations was performed immediately by the licensee during the survey. Areas requiring further decontamination were cleaned following the initial survey and, on November 7 and 8, 1983, ORAU personnel returned to the site to conduct followup measurements. Table 2 presents the results of these followup measurements. These results indicated that all areas were decontaminated to levels below the release guidelines.

Radionuclide Concentrations in Soil

Concentrations of radionuclides in soil from excavated areas are presented in Tables 3 and 4. Of eighteen soil samples from locations considered representative of the general trench excavations, two contained U-238 concentrations above the detection limits of the procedure (refer to Table 3). The levels of U-238 in these samples were 14.2 pCi/g (location 8) and 17.0 pCi/g (location 16). Only one of these samples (location 8) contained a significant level of U-235; the U-235 concentration in that sample was 13.2 pCi/g. Concentrations of other radionuclides were in the range of normal background soil.

Uranium concentrations in soil, collected at nine locations of elevated radiation identified during the surface scan, ranged from <4.25 to 9100 pCi/g

of U-238 and from 1.47 to 920 pCi/g of U-235 (see Table 4); no significant concentrations of other radionuclides were measured. Followup sampling was performed by ORAU during November 7 and 8, 1983 (see Table 5). Samples 19 and 24 were still well above the guidelines and additional cleanup was requested. Additional sampling of these two locations was performed on December 12, 1983. A small discrete area of contamination was identified by a radiation scan at location 19. This residual contamination, removed by sampling, contained U-238 and U-235 concentrations of 50.8 and 1.96 pCi/g respectively. Uranium 238 concentrations slightly above the limit were also detected at location 24; however, resampling performed at this location during January 17, 1984 did not disclose residual uranium levels above typical background concentrations.

## Comparison of Westinghouse and ORAU Instrumentation

Table 6 summarizes the instrumentation used by the licensee and ORAU for direct measurements of alpha and beta-gamma surface contamination. There were major differences in the detection principles and type of recording instruments used for alpha measurement. The Westinghouse instrument was a gas flow proportional counter with a lin/log count ratemeter; the ORAU instrument was a ZnS scintillator with a digital scaler. The Westinghouse equipment demonstrated a higher background and, consequently, a lower sensitivity than the ORAU equipment. Alpha measurements performed at the same locations indicated that ORAU results were consistently two to three times higher than those of Westinghouse. The differences were not resolved; however, ORAU's more conservative values confirm that the facility meets the NRC guidelines.

## SUMMARY

At the request of the Nuclear Regulatory Commission, ORAU conducted a radiological survey of the former Uranium Oxide Laboratory at the Westinghouse Nuclear Fuel Division in Cheswick, Pennsylvania. The purpose of the survey was to verify the adequacy and accuracy of the licensee's survey and to evaluate whether the facility satisfies NRC guidelines for release from licensing.

Contamination levels in some small isolated spots and several general areas were determined to be in excess of the established limits. Soil in excavated areas also exceeded release criteria. These findings indicated that additional cleanup was required.

Following further decontamination, surfaces and excavated areas were resurveyed. All contaminated areas, identified by the initial ORAU survey, have been removed or adequately cleaned, and the radiological status of the facility complies with the NRC guidelines for release for unrestricted use by the general public.

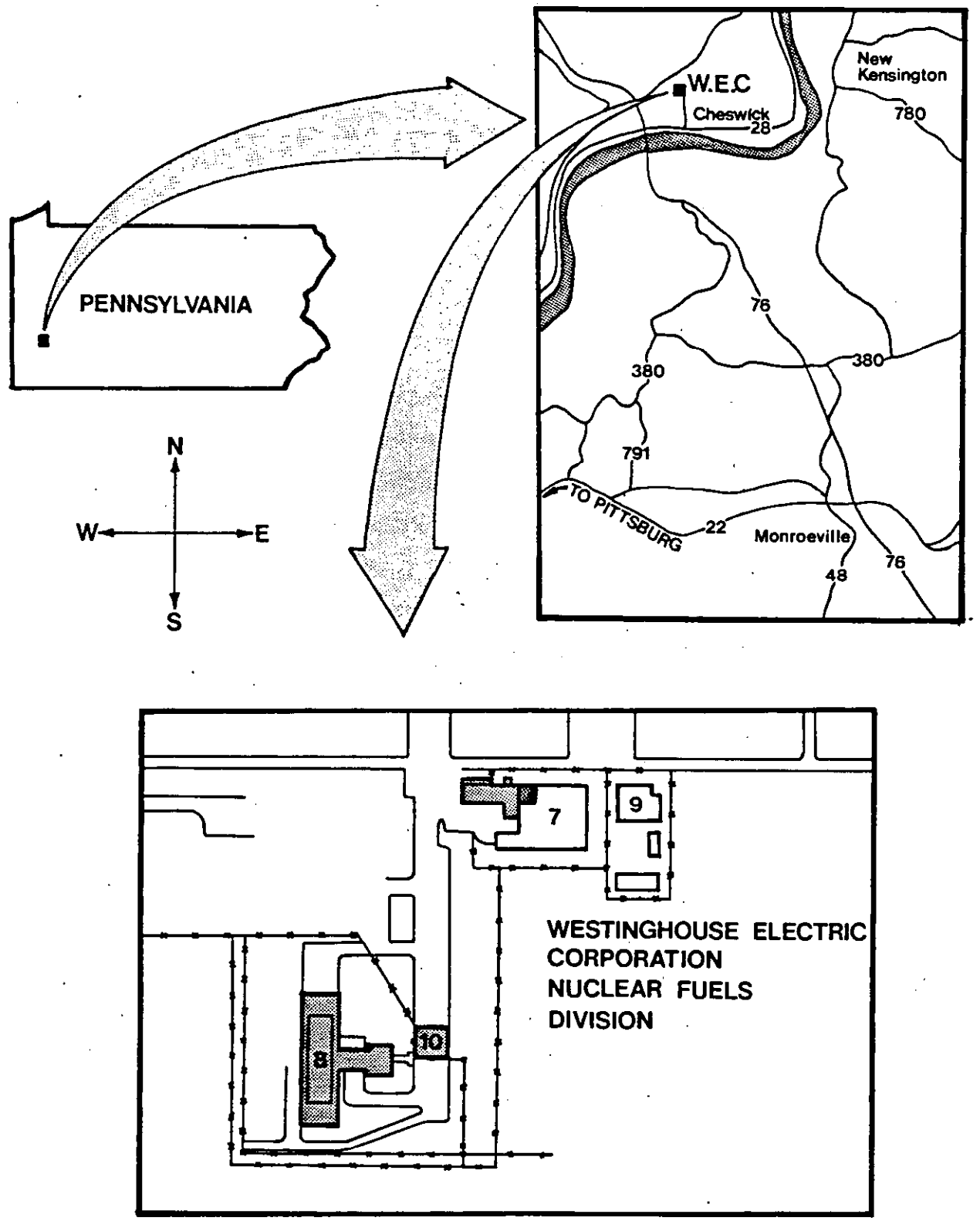
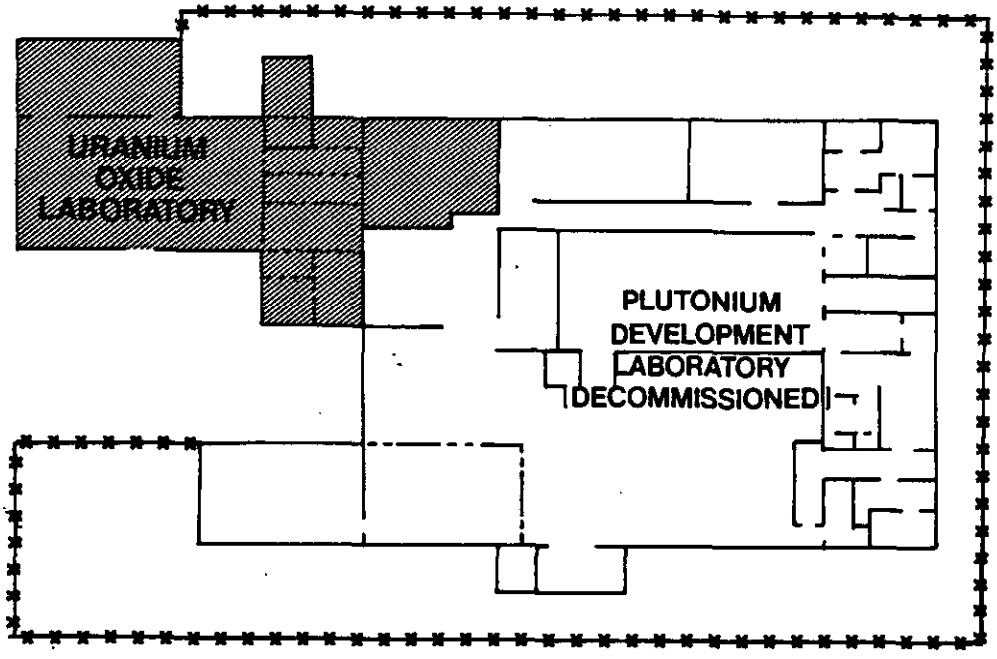



FIGURE 1: Maps Indicating the Location and General Layout of the Westinghouse Nuclear Fuels Division.



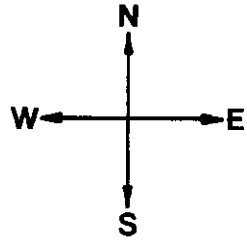




FIGURE 2: Plan of Building 7 Indicating the Uranium Oxide Laboratory.

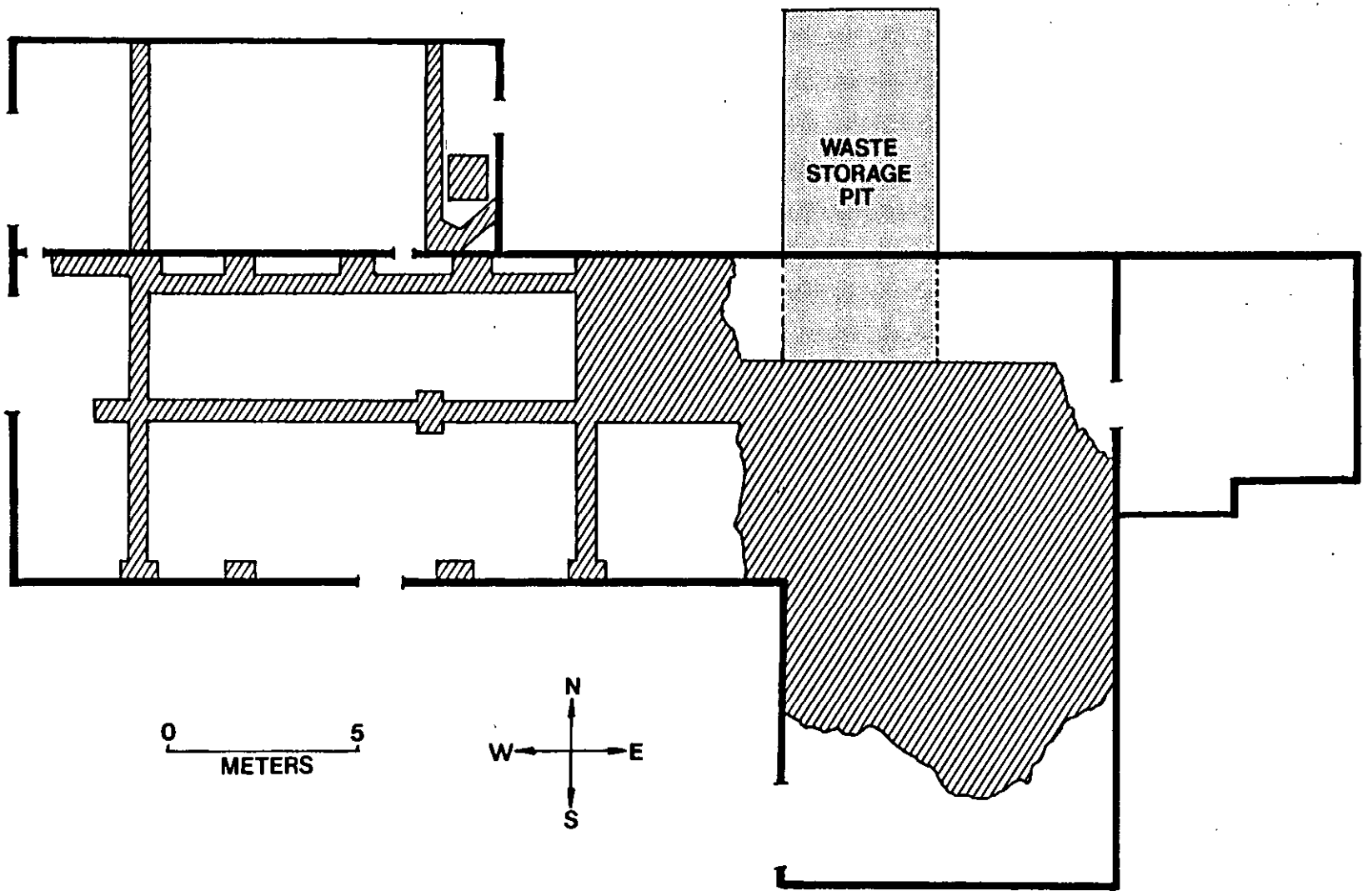



FIGURE 4: Floor Plan of the Uranium Oxide Laboratory with Interior Partitions Removed and Indicating (shaded) Excavated Trench Areas.

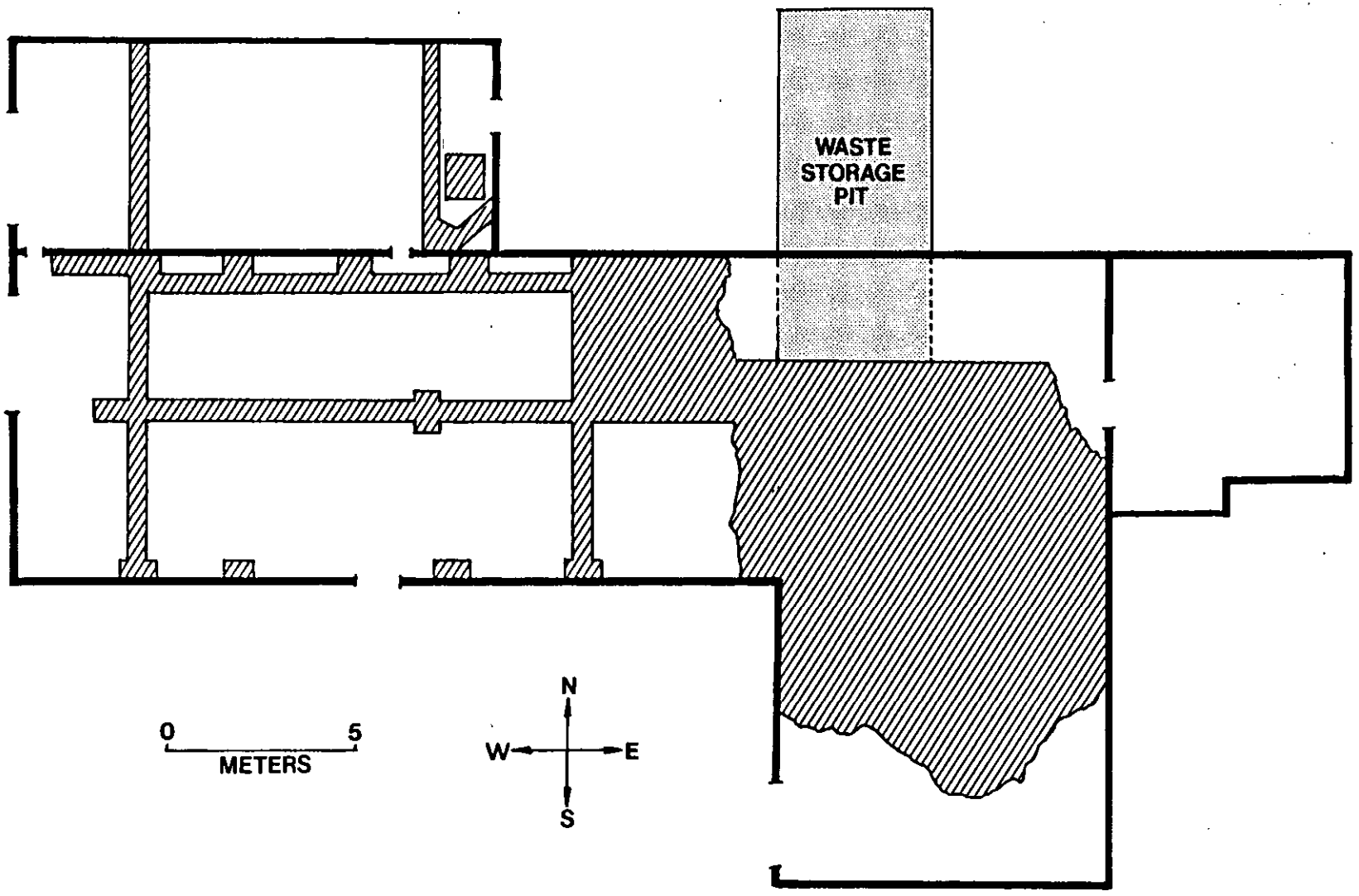



FIGURE 4: Floor Plan of the Uranium Oxide Laboratory with Interior Partitions Removed and Indicating (shaded) Excavated Trench Areas.

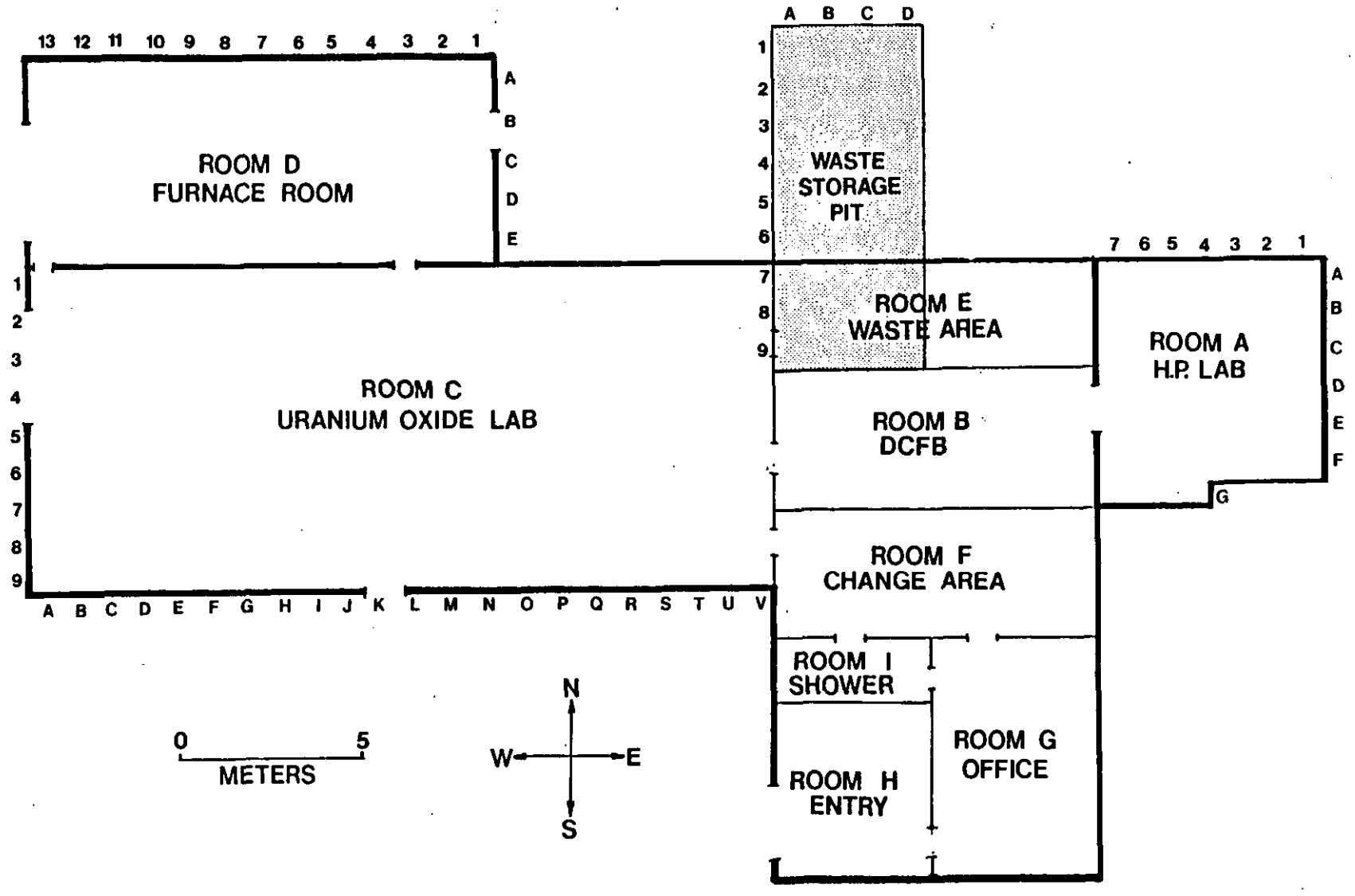



FIGURE 5: Grid System for Reference of Survey Measurements. (Grid is indicated only for rooms where residual contamination was identified by the ORAU survey.)

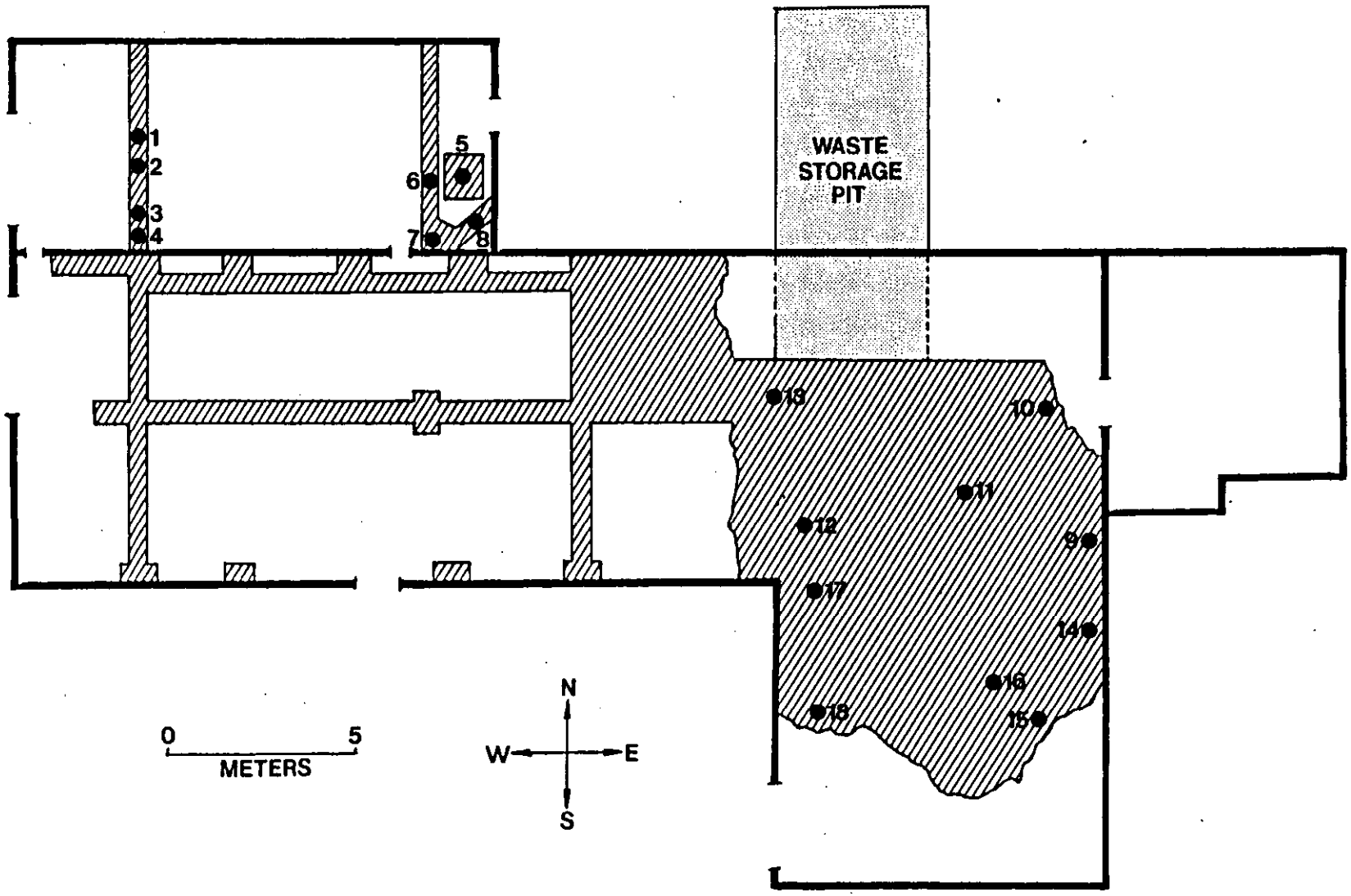



FIGURE 6: Locations of Soil Samples Collected to Provide Representative Coverage of Excavated Areas.

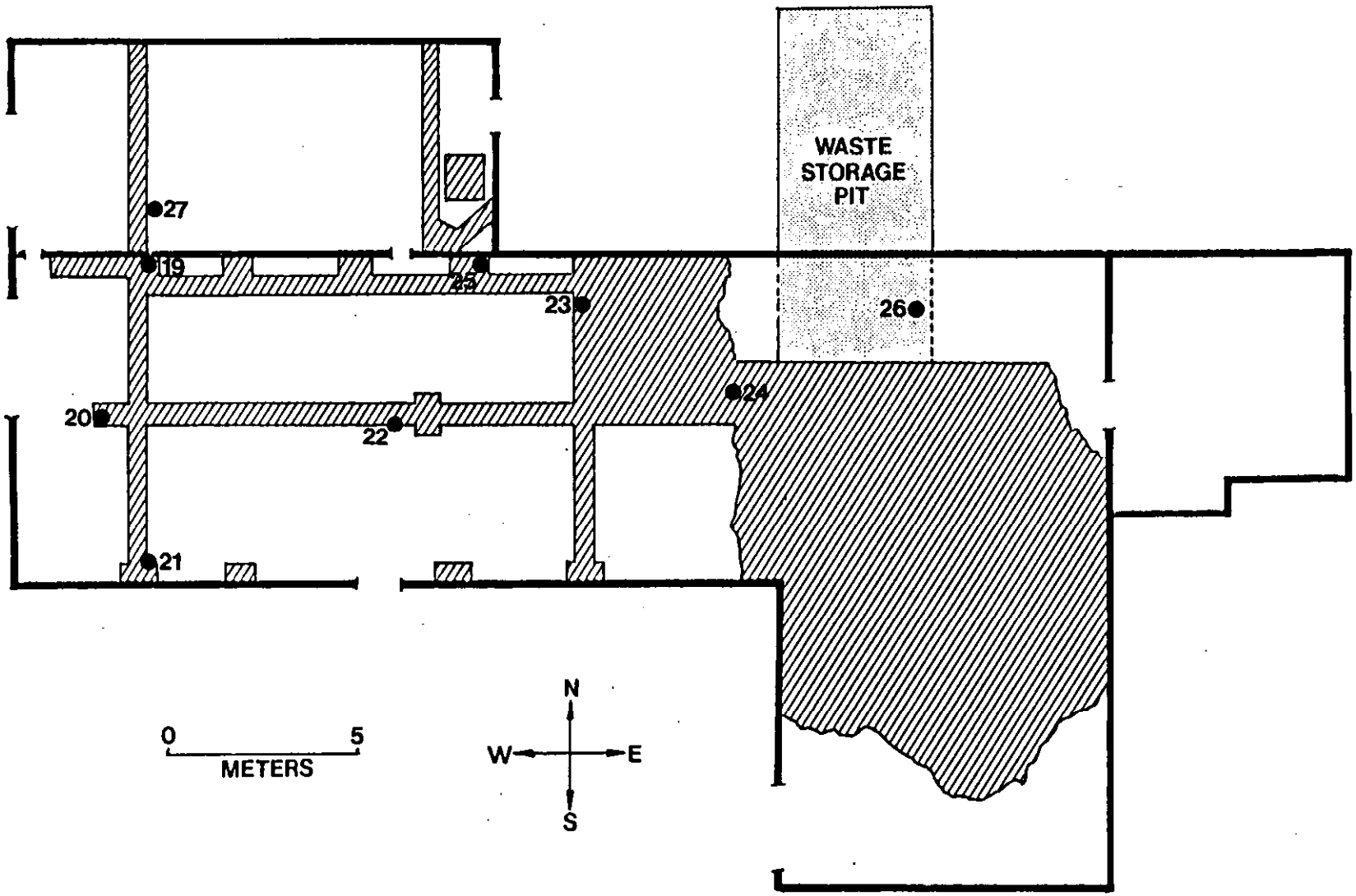



FIGURE 7: Locations of Soil Samples from Elevated Areas Identified by the Surface Scan.

TABLE 1

SUMMARY OF SURFACE CONTAMINATION MEASUREMENTS

| Room | Surface | No. of Blocks Surveyed | No. of Blocks Exceeding Criteria[a] | RANGE OF CONTAMINATION LEVELS | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | | Total Contamination (dpm/100 $cm^2$) | | Beta-Gamma Dose Rate (mrad/h) | Transferable Contamination (dpm/100 $cm^2$) | |
| | | | | Alpha | Beta-Gamma | | Alpha | Beta |
| A | Floor | 7 | 4 | <37 - 1,200 | <550 - 161,000 | <0.01 - 4.31 | <2 - 19 | <3 - 27 |
| | Walls | 12 | 0 | <37 - 286 | <550 - 1,087 | <0.01 - 0.03 | <2 - 24 | 4 - 59 |
| | Ceiling | 5 | 0 | <37 - 156 | <550 - 8,190 | <0.01 - 0.22 | <2 - 37 | 9 - 70 |
| C | Floor | 42 | 11 | 78 - 3,920 | <550 - 104,900 | <0.01 - 2.81 | <2 - 41 | 5 - 85 |
| | Walls | 26 | 0 | <37 - 130 | <550 - 659 | <0.01 - 0.02 | <2 - 3 | <3 - 8 |
| | Ceiling | 13 | 0 | <37 - 751 | <550 - 1,780 | <0.01 - 0.05 | <2 - 7 | <3 - 42 |
| D | Floor | 20 | 7 | 61 - 4,920 | <550 - 116,000 | <0.01 - 3.10 | <2 - 17 | 5 - 160 |
| | Walls | 17 | 0 | <37 - 139 | <550 - 926 | <0.01 - 0.03 | <2 | <3 - 8 |
| | Ceiling | 7 | 1 | <37 - 361 | <550 - 45,800 | <0.01 - 1.23 | <2 | 5 - 29 |
| E | Floor | 10 | 6 | 70 - 3,230 | <550 - 109,000 | <0.01 - 2.92 | <2 - 110 | 5 - 230 |
| | Walls | 5 | 0 | <37 - 445 | <550 - 659 | <0.01 - 0.02 | <2 - 10 | 4 - 19 |
| | Ceiling | 3 | 0 | <37 - 1,460 | <550 - 1,460 | <0.01 - 0.04 | 3 - 10 | 7 - 17 |
| F | Walls | 2 | 0 | 44 - 139 | <550 | <0.01 | <2 | 6 - 8 |
| | Ceiling | 2 | 0 | 39 - 195 | <550 - 553 | <0.01 - 0.01 | <2 - 4 | 10 |
| G | Floor | 2 | 0 | 95 - 334 | <550 - 820 | <0.01 - 0.02 | 3 - 4 | 4 - 11 |
| | Walls | 5 | 0 | 61 - 250 | <550 | <0.01 | <2 - 6 | 4 - 9 |
| | Ceiling | 1 | 0 | 44 - 111 | <550 | <0.01 | <2 | 7 |

TABLE 1, cont.

SUMMARY OF SURFACE CONTAMINATION MEASUREMENTS

| Room | Surface | No. of Blocks Surveyed | No. of Blocks Exceeding Criteria | RANGE OF CONTAMINATION LEVELS | | | |
|---|---|---|---|---|---|---|---|
| | | | | Total Contamination (dpm/100 $cm^2$) | | Beta-Gamma Dose Rate (mrad/h) | Transferable Contamination (dpm/100 $cm^2$) |
| | | | | Alpha | Beta-Gamma | | Alpha / Beta |
| H | Floor | 2 | 0 | 89 - 278 | <550 - 820 | <0.01 - 0.02 | <2 - 3 / <3 - 7 |
| | Walls | 4 | 0 | <37 - 14 | <550 - 659 | <0.01 - 0 02 | <2 / 4 - 7 |
| | Ceiling | 2 | 0 | <37 - 56 | <550 | <0.01 | <2 - 3 / 7 - 9 |
| PIT | Floor | 36 | 7 | <37 - 2,360 | <550 - 191,000 | <0.01 - 5.11 | <2 - 16 / 5 - 34 |
| | Walls | 48 | 5 | 39 - 3,253 | <550 - 24,200 | <0.01 - 0.65 | <2 - 13 / <3 - 33 |
| | Ceiling | 2 | 0 | 67 - 130 | <550 - 713 | <0.01 - 0.02 | <2 / 4 - 6 |

[a] NRC criteria is provided in Appendix C and discussed in the Survey Procedures section of the report.

TABLE 2

AREAS WHERE SURFACE CONTAMINATION EXCEED GUIDELINES[a]

| Location | | | Initial Survey | | | | | | After Decontamination | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | Total Contamination (dpm/100 $cm^2$) | | | | Transferable Contamination (dpm/100 $cm^2$) | | Total Contamination (dpm/100 $cm^2$) | |
| | | | Alpha | | Beta-Gamma | | | | Beta-Gamma | |
| Room | Surface | Grid ID | Avg. | Max. | Avg. | Max. | Alpha | Beta | Avg. | Max. |
| A | Floor | E-3 | 47 | 78 | 553 | 51,900 | <2 | 15 | 538 | 1,180 |
| | | E-7 | 95 | 156 | 3,790 | 44,000 | 9 | 10 | 470 | 616 |
| | | F-3 | 120 | 286 | 2,960 | 40,500 | 6 | 6 | 650 | 840 |
| | | G-5 | 78 | 1,220 | 3,150 | 161,000 | 19 | 27 | 2,220 | 2,630 |
| C | Floor | C-9 | 78 | 139 | 7,750 | 27,500 | 3 | 11 | 1,060 | 3,420 |
| | | E-9 | 868 | 2,990 | 6,460 | 19,500 | 13 | 23 | 3,630 | 7,340 |
| | | F-8 | 1,510 | 3,670 | 15,100 | 39,100 | 41 | 42 | 1,500 | 3,580 |
| | | G-8 | 934 | 1,950 | 10,140 | 41,500 | 34 | 85 | 1,460 | 3,080 |
| | | H-9 | 951 | 1,590 | 5,910 | 19,400 | 37 | 48 | 2,530 | 3,980 |
| | | J-4 | 228 | 306 | 8,410 | 39,500 | 6 | 13 | 661 | 1,120 |
| | | O-1 | 284 | 3,920 | 1,800 | 80,000 | 26 | 16 | 2,150 | 4,760 |
| | | P-1 | 1,410 | 3,590 | 19,700 | 89,800 | 17 | 34 | 4,540 | 3,810 |
| | | R-7 | 606 | 1,310 | 7,610 | 27,700 | <2 | 5 | 1,780 | 3,020 |
| | | T-1 | 612 | 1,330 | 9,390 | 45,100 | <2 | 16 | b | b |
| | | T-2 | 350 | 1,080 | 21,600 | 105,000 | 13 | 18 | b | b |
| D | Floor | B-2 | 1,300 | 4,170 | 23,300 | 116,000 | 130 | 160 | 1,470 | 1,680 |
| | | B-12 | 1,220 | 1,700 | 8,940 | 33,700 | 10 | 22 | 3,750 | 4,980 |
| | | C-2 | 2,980 | 4,920 | 13,000 | 33,500 | 3 | 10 | 1,660 | 6,780 |
| | | C-12 | 1,220 | 1,890 | 10,700 | 94,200 | <2 | 10 | 756 | 8,120 |
| | | D-3 | 589 | 1,470 | 9,870 | 21,100 | 6 | 18 | 2,707 | 5,880 |
| | | D-12 | 146 | 222 | 9,670 | 93,300 | 6 | 19 | <550 | 952 |

TABLE 2, cont.

AREAS WHERE SURFACE CONTAMINATION EXCEEDED GUIDELINES

| Location | | | Initial Survey | | | | | | After Decontamination | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | Total Contamination (dpm/100 $cm^2$) | | | | Transferable Contamination (dpm/100 $cm^2$) | | Total Contamination (dpm/100 $cm^2$) | |
| | | | Alpha | | Beta-Gamma | | | | Beta-Gamma | |
| Room | Surface | Grid ID | Avg. | Max. | Avg. | Max. | Alpha | Beta | Avg. | Max. |
| D | Floor | E-5 | 83 | 111 | 5,480 | 13,700 | <2 | 6 | 2,260 | 9,630 |
| | Ceiling | B-5 | 111 | 195 | <550 | 45,800 | 13 | 29 | c | c |
| E | Floor | A-4 | 423 | 612 | 2,660 | 15,900 | 14 | 41 | Metal plates composing the floor for Room E were removed and sent to authorized brial. | |
| | | A-5 | 600 | 1,000 | 6,830 | 109,000 | 21 | 37 | | |
| | | A-6 | 773 | 1,170 | 5,500 | 105,000 | 11 | 56 | | |
| | | A-7 | 901 | 3,230 | 3,400 | 30,000 | 14 | 43 | | |
| | | B-4 | 589 | 862 | 3,450 | 71,100 | 110 | 230 | | |
| | | C-4 | 1,050 | 1,560 | 9,110 | 31,300 | 7 | 7 | | |
| Pit | Floor | A-9 | 621 | 676 | 22,700 | 44,00 | 7 | 19 | 1,770 | 5,420 |
| | | B-7 | 461 | 858 | 5,420 | 10,700 | 16 | 26 | 1,090 | 1,790 |
| | | C-7 | 450 | 1,330 | 14,900 | 39,100 | 3 | 9 | 1,410 | 9,520 |
| | | D-3 | 217 | 208 | 4,370 | 17,700 | | | 997 | 2,860 |
| | | D-4 | 161 | 208 | 4,780 | 15,400 | | | 2,220 | 5,380 |
| | | D-5 | 172 | 338 | 2,500 | 27,400 | | | 2,530 | 4,420 |
| | | D-7 | | 2,360 | | 191,000 | 10 | 26 | 2,340 | 5,710 |
| | West Wall | I-3 | 1,150 | 1,120 | 3,490 | 7,120 | 19 | 18 | <550 | 840 |
| | | I-4 | 645 | 598 | 5,350 | 15,900 | 160 | 150 | 644 | 1,010 |

TABLE 2, cont.

AREAS WHERE SURFACE CONTAMINATION EXCEEDED GUIDELINES

| Location | | | Initial Survey | | | | | | After Decontamination | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | Total Contamination (dpm/100 $cm^2$) | | | | Transferable Contamination (dpm/100 $cm^2$) | | Total Contamination (dpm/100 $cm^2$) | |
| | | | Alpha | | Beta-Gamma | | | | Beta-Gamma | |
| Room | Surface | Grid ID | Avg. | Max. | Avg. | Max. | Alpha | Beta | Avg. | Max. |
| Pit | South Wall | C-4 | 1,130 | 1,980 | 7,060 | 17,400 | <2 | 15 | 4,050 | 4,980 |
| | | D-1 | 117 | 156 | 194 | 24,200 | <2 | 8 | <550 | <550 |
| | | D-4 | 2,490 | 3,250 | 5,570 | 7,070 | 13 | 22 | <550 | 1,120 |

[a] NRC criteria is provided in Appendix C and discussed in the Survey Procedures section of the report.

[b] Follow-up measurements were not obtainable, floor slab was removed at this location.

[c] Follow-up measurements were not obtainable, portion of ceiling where contamination was identified was removed.

TABLE 3

RADIONUCLIDE CONCENTRATIONS IN SOIL COLLECTED
RANDOMLY FROM EXCAVATED AREAS

| Location[a] No. | Radionuclide Concentrations (pCi/g) | | | | | |
|---|---|---|---|---|---|---|
| | U-238 | U-235 | Th-228 | Th-232 | Ra-226 | K-40 |
| 1 | <3.95 | <0.25 | 0.90 ± 0.28 | 1.35 ± 0.47 | 1.21 ± 0.27 | 15.3 ± 2.3 |
| 2 | <2.29 | <0.14 | 0.87 ± 0.25 | 1.13 ± 0.37 | 0.99 ± 0.16 | 12.4 ± 1.8 |
| 3 | <4.76 | <0.23 | 1.24 ± 0.27 | 0.90 ± 0.56 | 1.04 ± 0.26 | 13.5 ± 2.2 |
| 4 | <4.94 | <0.17 | 1.12 ± 0.27 | 0.87 ± 0.36 | 1.06 ± 0.25 | 15.9 ± 2.1 |
| 5 | <5.32 | 0.33 ± 0.42 | 0.91 ± 0.31 | 1.52 ± 0.64 | 1.00 ± 0.23 | 11.9 ± 2.0 |
| 6 | <4.91 | 0.33 ± 0.47 | 0.92 ± 0.27 | 1.17 ± 0.35 | 1.00 ± 0.18 | 14.3 ± 2.2 |
| 7 | <4.92 | <0.25 | 1.10 ± 0.33 | 1.38 ± 0.58 | 1.04 ± 0.28 | 13.8 ± 2.3 |
| 8 | 14.2 ± 8.0[b] | 13.2 ± 0.8 | 0.90 ± 0.34 | 0.96 ± 0.45 | 1.01 ± 0.21 | 11.8 ± 1.9 |
| 9 | <6.10 | <0.33 | 1.18 ± 0.30 | 1.33 ± 0.36 | 0.93 ± 0.17 | 14.9 ± 1.9 |
| 10 | <4.17 | <0.22 | 1.01 ± 0.39 | 1.28 ± 0.40 | 1.20 ± 0.26 | 16.8 ± 2.0 |
| 11 | <5.65 | <0.30 | 1.00 ± 0.35 | 0.65 ± 0.44 | 0.81 ± 0.16 | 8.49 ± 1.6 |
| 12 | <4.69 | <0.16 | 0.65 ± 0.24 | 1.06 ± 0.37 | 0.77 ± 0.16 | 10.2 ± 1.8 |
| 13 | <1.37 | 0.09 | 0.63 ± 0.35 | 0.61 ± 0.31 | 0.86 ± 0.21 | 9.06 ± 1.7 |
| 14 | <5.47 | <0.31 | <0.21 | 0.83 ± 0.28 | 0.92 ± 0.20 | 12.3 ± 1.8 |
| 15 | <2.75 | <0.16 | 0.88 ± 0.22 | 1.10 ± 0.33 | 0.88 ± 0.26 | 13.7 ± 2.0 |
| 16 | 17.0 ± 8.7 | <0.18 | 0.90 ± 0.31 | 0.96 ± 0.31 | 0.84 ± 0.22 | 13.1 ± 2.1 |
| 17 | <4.59 | <0.29 | 0.84 ± 0.24 | 1.18 ± 0.34 | 0.82 ± 0.15 | 13.5 ± 2.0 |
| 18 | <4.01 | <0.15 | 0.70 ± 0.19 | 0.91 ± 0.27 | 0.87 ± 0.20 | 14.2 ± 2.1 |

[a] Locations are identified on Figure 6.
[b] Errors are 2σ based on counting statistics.

TABLE 4

RADIONUCLIDE CONCENTRATIONS IN SOIL COLLECTED FROM LOCATIONS IDENTIFIED BY SCANNING EXCAVATED AREAS OF BUILDING 7

| Location[a] No. | Radionuclide Concentrations (pCi/g) | | | | | |
|---|---|---|---|---|---|---|
| | U-238 | U-235 | Th-238 | Th-232 | Ra-226 | K-40 |
| 19 | 1,390 ± 57[b] | 47.0 ± 2.6 | 1.42 ± 0.64 | 0.96 ± 0.22 | 0.89 ± 0.52 | 9.76 ± 2.30 |
| 20 | <4.25 | 1.47 ± 0.50 | 1.02 ± 0.39 | 1.24 ± 0.32 | 1.30 ± 0.21 | 14.7 ± 2.1 |
| 21 | 119 ± 21 | 4.02 ± 1.00 | 1.21 ± 0.33 | 1.37 ± 0.50 | 1.20 ± 0.32 | 14.4 ± 2.3 |
| 22 | 162 ± 25 | 8.29 ± 1.19 | 1.43 ± 0.48 | 1.13 ± 0.67 | 1.29 ± 0.22 | 17.3 ± 2.5 |
| 23 | 9,100 ± 200 | 919 ± 8 | <0.91 | <0.70 | 1.35 ± 1.21 | 5.49 ± 2.9 |
| 24 | 1,990 ± 78 | 99.1 ± 2.7 | <0.42 | 0.98 ± 0.64 | 1.05 ± 0.54 | 8.61 ± 2.22 |
| 25 | 189 ± 24 | 36.4 ± 2.0 | 0.94 ± 0.31 | 0.77 ± 0.48 | 0.87 ± 0.23 | 10.7 ± 1.8 |
| 26[c] | 309 ± 27 | 22.9 ± 1.1 | <0.14 | <0.10 | 2.24 ± 0.13 | 1.04 ± 0.62 |
| 27[d] | 98.7 ± 21.0 | 3.29 ± 0.57 | 0.79 ± 0.38 | 1.32 ± 0.36 | 1.18 ± 0.32 | 12.9 ± 2.6 |

[a] Locations are identified on Figure 7.

[b] Errors are 2σ based on counting statistics.

[c] Sediment from the sump in the waste storage pit.

[d] Collected below the concrete floor slab along east side of trench.

TABLE 5

SUMMARY OF FOLLOWUP MEASUREMENTS OF
RADIONUCLIDES IN SOIL

| Location[a] No. | Radionuclide Concentrations (pCi/g)[b] | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | INITIAL MEASUREMENT | | AFTER FIRST DECONTAMINATION | | AFTER SECOND DECONTAMINATION | | AFTER THIRD CONTAMINATION | |
| | U-238 | U-235 | U-238 | U-235 | U-238 | U-235 | U-238 | U-235 |
| 19 | 1,390 ± 57[c] | 47.0 ± 2.6 | 353 ± 34 | 8.59 ± 1.41 | 50. 8 ± 3.1 | 1.96 ± 0.80 | d | d |
| 21 | 119 ± 21 | 4.02 ± 1.00 | <4.81 | <0.45 | --- | --- | --- | --- |
| 22 | 162 ± 25 | 8.29 ± 1.19 | <6.22 | <0.50 | --- | --- | --- | --- |
| 23 | 9,100 ± 200 | 919 ± 8 | <14.6 | <0.35 | --- | --- | --- | --- |
| 24 | 1,990 ± 78 | 99.1 ± 2.7 | 109 ± 19 | 5.78 ± 1.07 | 39.5 ± 2.4 | 2.22 ± 0.64 | <2.49 | 0.23 ± 0.26 |
| 25 | 189 ± 25 | 36.4 ± 2.0 | 36.8 ± 15.7 | 0.84 ± 0.75 | --- | --- | --- | --- |
| 26 | 309 ± 27 | 22.9 ± 1.1 | d | d | --- | --- | --- | --- |
| 27 | 98.7 ± 21 | 3.29 ± 0.57 | <5.72 | 5.31 ± 0.88 | --- | --- | --- | --- |

[a] Locations are identified on Figure 7.
[b] Soil contamination criteria for total uranium is 35 pCi/g.
[c] Errors are 2σ based on counting statistics.
[d] All contaminated residue was removed by the licensee.

TABLE 6

COMPARISON OF SURVEY INSTURMENTS USED BY WESTINGHOUSE AND ORAU

| WESTINGHOUSE | | | | ORAU | | | |
|---|---|---|---|---|---|---|---|
| Instrument | Active Area | Correction Factor (CPM to dpm/100 $cm^2$) | MDA | Instrument | Active Area | Correction Factor (CPM to dpm/100 $cm^2$) | MDA |
| **Total Alpha Contamination Survey** | | | | **Total Alpha Contamination Survey** | | | |
| PAC-4G (Eberline) ratemeter with a gas flow proportional detector (0.85 mg/$cm^2$ Mylar window) | 50 $cm^2$ | 4 | 50 dpm | PRS-1 (Eberline) scaler/ratemeter with Model AC3-7 ZnS Alpha Scintillation detector | 59 $cm^2$ | 13.9 | 22 dpm |
| **Beta-Gamma Contamination Survey** | | | | **Beta-Gamma Contamination Survey** | | | |
| PRS-1 (Eberline) scaler/ratemeter with Model HP-190 end-window G-M detector (1.4-2.0 mg/$cm^2$ Mica window) | 6.6 $cm^2$ | 84 | 167 dpm | PRS-1 (Eberline) scaler/ratemeter with Model HP-260 Pancake G-M detector (1.8 mg/$cm^2$ Mica window) | 15 $cm^2$ | 26.7 | 82 dpm |

# APPENDIX A

## MAJOR ANALYTICAL EQUIPMENT

APPENDIX A

Major Analytical Equipment

The display or description of a specific product is not to be construed and an endorsement of that product or its manufacturer by the authors or their employer.

A. Direct Radiation Measurements

Eberline "RASCAL"
Portable Scaler/Ratemeter
Model PRS-1
(Eberline, Sante Fe, NM)

Eberline PRM-6
Portable Ratemeter
(Eberline, Sante Fe, NM)

Eberline Beta-Gamma "Pancake" Probe
Model HP-260
(Eberline, Sante Fe, NM)

Eberline Alpha Scintillation Probe
Models AC-3-7
(Eberline, Sante Fe, NM)

Victoreen Gamma Scintillation (NaI) Probe
Model 489-55
(Victoreen, Inc., Cleveland, OH)

Eberline Low-Energy Gamma Scintillation Probe
Model PG-2
(Eberline, Sante Fe, NM)

Reuter-Stokes Pressurized Ionization Chamber
Model RSS-111
(Reuter-Stokes, Cleveland, OH)

B. Laboratory Analyses

Low Background Alpha-Beta Counter
Model LB5100-2080
(Tennelec, Inc., Oak Ridge, TN)

Ge(Li) Detectors (2)
Model LGCC2220SD, 23% efficiency
(Princeton Gamma-Tech, Princeton, NJ)

Used in conjunction with:
Lead Shield, SPG-16
(Applied Physical Technology, Smyrna, GA)

High-Purity Germanium Detector
Model GMX-23195-S, 23% efficiency
(EG&G ORTEC, Oak Ridge, TN)

Used in conjunction with:
Lead Shield, G-16
(Gamma Products Inc., Palos Hills, IL)

Pulse Height Analyzer, ND680
Model 88-0629
(Nuclear Data, Inc., Schaumburg, IL)

# APPENDIX B

# ANALYTICAL PROCEDURES

Appendix B

Analytical Procedures

## Alpha and Beta-Gamma Measurements

Measurements of direct beta-gamma radiation levels were performed using Eberline "Rascal" Model PRS-1 portable scaler/ratemeters with Model HP-260 thin-window "pancake" G-M probes. Measurements of direct alpha radiation levels were performed using Eberline Model PRS-1 portable ratemeter/scalers with Model AC-3-7 ZnS alpha scintillation probes. Count rates (cpm) were converted to disintegration rates (dpm/100 $cm^2$) by dividing the net rate by the $4\pi$ efficency and correcting for active area of the detector.

The average background count rate was 38 cpm for the "pancake" G-M probes. Background count rates for the ZnS alpha probes averaged approximately 1 cpm; however, a conservative value of 0 cpm was employed for the calculations. Effective window areas were 15 $cm^2$ for the G-M probes and 59 $cm^2$ for the ZnS probes. Using this technique, the count rates recorded by the detector are converted to contamination levels as if the distribution were constant over a 100 $cm^2$ area. This conservatively overestimates disintegration rates for small areas.

## Surface Scans

Surface scans were performed by passing the probe slowly (20-30 cm/s) over the surface. The distance between the probe and the surface was maintained at a minimum - nominally about 1 cm. Identification of elevated levels was based on increases in the audible signal from the recording or indicating instrument. Combinations of detectors and instruments for the scans were:

Gamma - NaI Scintillation probe with PRM-6 ratemeter
Low-Energy Gamma - Thin NaI Scintillation probe with PRM-6 ratemeter
Beta-Gamma - Pancake G-M probe with "Rascal" scaler/ratemeter
Alpha - ZnS probe with "Rascal" scaler/ratemeter.

## Beta-Gamma Dose Rate Measurements

Beta and gamma dose rates were calculated individually and the results summed for a combined beta-gamma dose rate. Beta dose rates were calculated by applying the conversion factor of 1400 cpm/mrad/h to the net beta count rate. The gamma dose rate component was assumed to be a constant 0.013 mrad/h, based on an average exposure rate of 13 μR/h measured in the facility.

## Gamma Exposure Rate

Measurements of gamma exposure rates were performed using a Reuter-Stokes pressurized ionization chamber. The chamber was placed at 1 m above the floor at various locations throughout the facility and the average of several readings at each location was determined.

## Transferable Contamination Measurements

Smear measurements were performed on numbered filter paper disks, 47 mm in diameter. Each smear was sealed in an labeled envelope with the location and other pertinent information recorded. A low background alpha-beta counting system was used to count individual smears.

## Soil and Residue Sample Analysis

Samples were dried at 120° C, finely ground, mixed, and a portion placed in a 0.5 liter Marinelli beaker. The quantity placed in each beaker was chosen to reproduce the calibrated counting geometry and typically ranged from 500 to 800 g of soil. Net weights were determined and the samples counted using either Ge(Li) high purity germanium detectors coupled to a Nuclear Data Model ND 680 pulse height analyzer. The following energy peaks were used for determination of the radionuclides of concern:

U-235 - 0.143 MeV
U-238 - 1.001 MeV from Pa-234m (secular equilibrium assumed)
Th-228 - 0.583 MeV from Tl-208 (secular equilibrium assumed)
Th-232 - 0.911 MeV from Ac-228 (secular equilibrium assumed)

Ra-226 - 0.609 MeV from Bi-214 (corrected for equilibrium conditions)
K-40 - 1.460 MeV

Peak identification and calculations of concentrations, statistical errors, and detection sensitivities were performed by the computer capabilities inherent in the analyzer system.

## Errors and Detection Limits

The errors associated with the analytical data presented in the tables of this report, represent the 95% ($2\sigma$) confidence levels for that data. These errors were calculated, based on both the gross sample count levels and the associated background count levels. When the net sample count was less than the $2\sigma$ statistical deviation of the background count, the sample concentration was reported as less than the minimum detectable activity (<MDA). This means that the radionuclide was not present, to the best of our ability to measure it, utilizing the analytical techniques described in this appendix. Because of variations in background levels, caused by other constituants in the samples, the MDAs for specific radionuclides differ from sample to sample.

## Calibration and Quality Assurance

Portable survey equipment and laboratory and analytical instruments were calibrated using NBS-traceable standards. Quality control procedures on all instruments included daily background and check-source measurements to confirm equipment operation within acceptable statistical fluctuations. The ORAU laboratory participates in the EPA Quality Assurance Program.

# APPENDIX C

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

# 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
Divison of Fuel Cycle & Material Safety
Washington, D.C. 20555

July 1982

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

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

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

3. The radioactivity on the interior surfaces of pipes, drain lines, or 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 or premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

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

   a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

   b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of

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

a. Identify the premises.

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

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

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

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

TABLE C-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 $\alpha$/100 $cm^2$ | 15,000 dpm $\alpha$/100 $cm^2$ | 1,000 dpm $\alpha$/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 | 1000 dpm/100 $cm^2$ | 3000 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 others noted above. | 5000 dpm $\beta\gamma$/100 $cm^2$ | 15,000 dpm $\beta\gamma$/100 $cm^2$ | 1000 dpm $\beta\gamma$/100 $cm^2$ |

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

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

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

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

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

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