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Radiological Survey of the Former Bridgeport Brass Company Special Metals Extrusion Plant, Adrian, Michigan

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HEALTH AND SAFETY RESEARCH DIVISION
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Health and Safety Research Division

Final Report

RADIOLOGICAL SURVEY OF THE FORMER BRIDGEPORT BRASS COMPANY SPECIAL METALS EXTRUSION PLANT, ADRIAN, MICHIGAN

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ABSTRACT

A radiological survey was made of the former Bridgeport Brass Special Metals Extrusion Plant in Adrian, Michigan, now owned by General Motors Corporation. This plant was operated to extrude uranium metal which was used in the fabrication of reactor fuel for the Hanford, Washington, and Savannah River, South Carolina, plants. Activities at the Adrian plant included preparation of material for extrusion, abrasive sawing, storing, packaging, and shipping. When the original contract was concluded, most of the equipment was dismantled and salvaged. The current property owner cleaned much of the building and conducted his own radiological survey. The results of the General Motors survey indicated that the area originally involved in the uranium handling and processing operation was within tolerances under the provision of guidelines applicable at the time the facility was decommissioned. A comprehensive survey was conducted in that area by a team of health physicists from the Oak Ridge National Laboratory (ORNL). The results of this survey tend to confirm the findings of the General Motors report, except that some floor areas were contaminated in excess of applicable guidelines and some off-gas ducts which had been used in the cutting area were found to be contaminated with uranium. These ducts were removed, the floor areas were cleaned, and a subsequent resurvey of the plant was made by ORNL during February and March, 1977.

In April, 1979, an additional survey of a portion of the facility was conducted by ORNL health physicists after learning that service pits had existed beneath the extrusion units. Sometime after extrusion operations ceased, these pits were filled with sand and covered over at the existing floor level with concrete. Results of this survey revealed concentrations of ^{238}U up to 21,000 pCi/g of residue, scale, and other miscellaneous materials collected from the bottom of service pits, service manholes, and holding tanks.

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INTRODUCTION

During the 1950s, the Bridgeport Brass Company, a division of National Distillers and Chemical Corporation, operated a Special Metals Extrusion Plant in Adrian, Michigan, under contract AT-(30-1)-1405 with the Department of Energy (DOE, then the Atomic Energy Commission [AEC]). The product of this operation was material for uranium fuel elements for reactors in Hanford, Washington, and at the Savannah River Plant in South Carolina. Uranium handled in this operation included depleted, natural, and up to 2.1% enriched.[1] There are no descriptive records available which indicate the full nature of the operations at this plant.

At the completion of work by the Bridgeport Brass Company, one large extrusion press was shipped to Reactive Metals, Inc., in Ashtabula, Ohio, and put into operation there. Other equipment was dismantled and scrapped. The whereabouts of this material is unknown. The plant was sold to Martin-Marietta in the early 1960s. It was used by that company until 1974, when it was sold to General Motors, Chevrolet Manufacturing Division. Therefore, from about 1961 until the present, no records exist to document alpha and beta-gamma contamination levels on the floor, walls, fixtures, and structural members of the building.

In May, 1976, newspapers throughout the country carried articles pertaining to a report from the General Accounting Office recommending that DOE (then the Energy Research and Development Administration [ERDA]) expedite completion of radiological surveys at numerous sites throughout the U. S. in order to certify that no potential hazards existed at these sites. When General Motors officials learned through this release that the Adrian property was included in those under consideration, a decision was made to perform an in-house survey of the facility in order to determine the need for decontamination. Residual uranium was found in numerous places, especially on elevated horizontal surfaces, fixtures, and in floor cracks. The building was decontaminated using a vacuum cleaner with filtered exhaust. This cleaning operation netted eight 55-gal drums containing 1 ton of dust and dirt. According to an analysis of the residue, there was a total of 5 to 6 kg of uranium in the collected

dust. A follow-up survey was made by General Motors. Results suggested that the areas actually surveyed were within recognized tolerances. The Department of Energy (then ERDA) was then requested to confirm the General Motors results with a formal survey and to verify that the area met current guidelines for unrestricted release of property. In order to verify the results obtained by General Motors and to obtain an independent evaluation, ORNL was asked to inspect the facility, make exploratory measurements, and conduct a formal survey if one was required.

On August 9, 1976, a presurvey visit to the Adrian plant was made by H. W. Dickson of ORNL's Health and Safety Research Division (then Health Physics Division) and W. T. Thornton of DOE (then ERDA), Oak Ridge Operations (ORO). During this initial site visit, preliminary measurements were made in areas where particulate matter was apt to accumulate. It was found that most of the activity on the floor was confined to cracks in the concrete. There were a number of places where concrete had been chipped during cleanup to remove contaminated sections of the floor. Exploratory measurements revealed residual radioactivity, but the readings were less than 1000 α dpm/100 cm^2 by direct reading on the floor and less than 0.2 mrad/h at 1 cm from the floor due to beta-gamma radiation. In areas well above the floor, some alpha radioactivity was found on structural steel members but did not exceed 2,000 dpm/100 cm^2.

Because residual radioactivity approaching limiting average levels was detected during this visit, it was decided that a comprehensive survey of the building would be needed in order to assure the appropriateness of unrestricted use of the property. Based on the results of this preliminary visit, a plan was developed for the formal survey of this facility. The plan was approved by DOE (then ERDA), and the initial survey work was conducted during the period August 17-19, 1976.

DESCRIPTION OF PLANT AREA

The former Bridgeport Brass Company plant, only a portion of which was used for the uranium extrusion operations, is a large complex covering approximately 757,000 ft^2 (17.4 acres). A layout of the building was sketched from an original drawing and is given in Fig. 1. Metal extrusion, cutting, etc., was carried out in Bays 5 through 7, shown as the shaded area in Fig. 1. This area is approximately 40,800 ft^2, with a ceiling height which varies from 45 to 55 ft. Lighting was provided by several rows of fluorescent fixtures and by sunlight through windows in two 10-ft-high "monitors"* in Bays 5 and 7. These windows were opened and closed automatically by motorized actuators. The large open areas of this structure are afforded by a massive steel framework. Supported from this framing are crane rails, roof drain lines, electrical wires and conduits, water pipes, space heaters, and off-gas ducts. Airborne radioactive material which settled onto the horizontal surfaces of this network of material presented a difficult situation both with regard to cleanup and monitoring. The largest single surface was the concrete floor. Most of the remaining surfaces were concrete block walls and steel supports with the exception of a wood catwalk (45 ft above the floor) which traversed two sides and the north end of both the west monitor located in Bay 5 and the east monitor located in Bay 7.

A general view of the operational area during the AEC contract is shown in Fig. 2 and includes a nominal 3,800-ton-capacity press (lower right), saw (lower left), and pickling vats (upper left). A close-up view of the pickling vats and associated off-gas system is shown in Fig. 3. Blowers for numerous off-gas ducts in the exhaust system were located on the roof. Air from this off-gas system was released at some unknown height above the roof.

At the time of the survey, conducted in August, 1976, as well as at the time of the subsequent survey conducted in February-March, 1977, there was a surrounding concrete block wall on the east and south sides of

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*Monitors referred to here are raised sections of the roof which contain rows of windows for lighting.

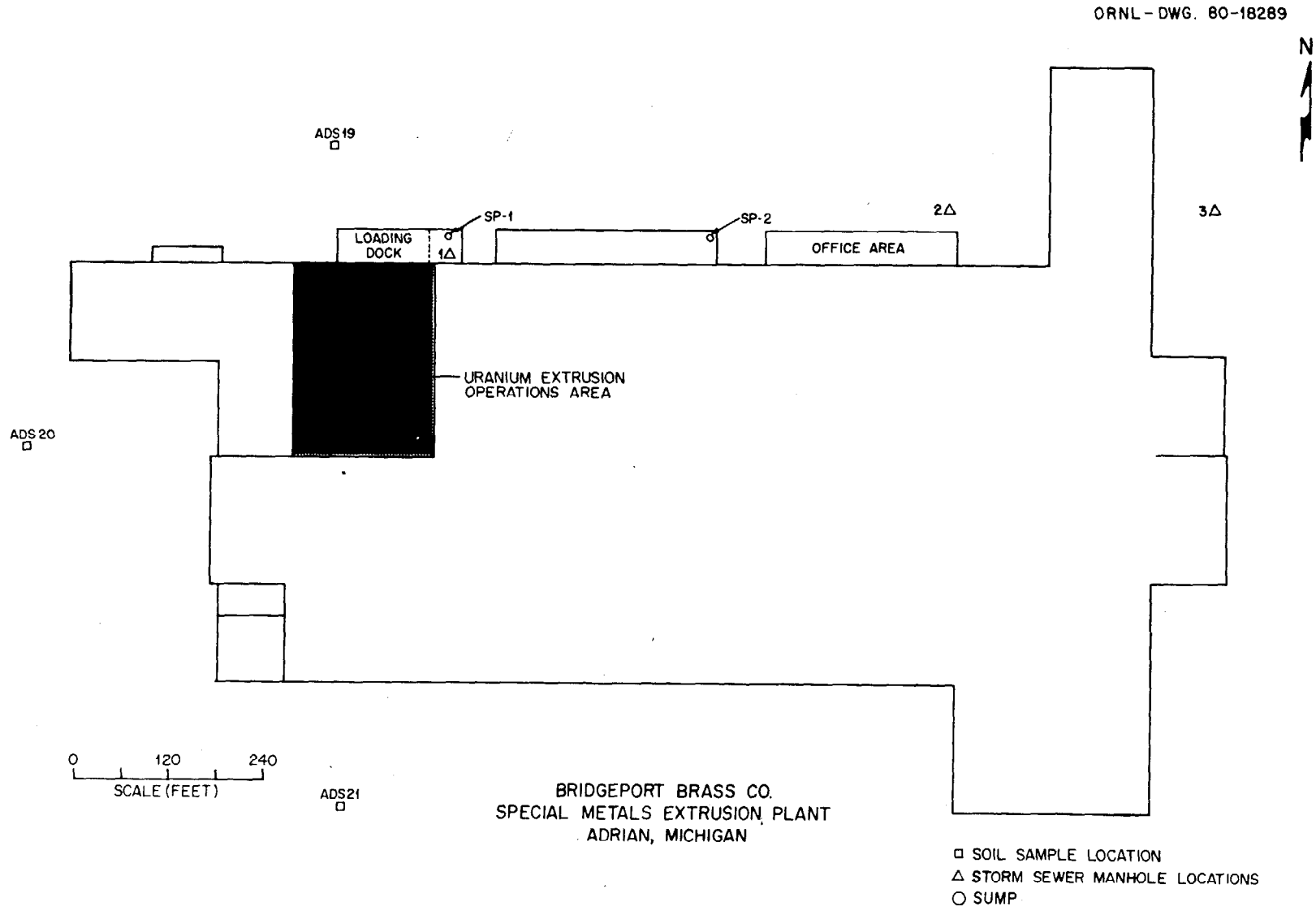

Fig. 1. Plan view of former Bridgeport Brass Special Metals Extrusion Plant.

ORNL-Photo 1192-79

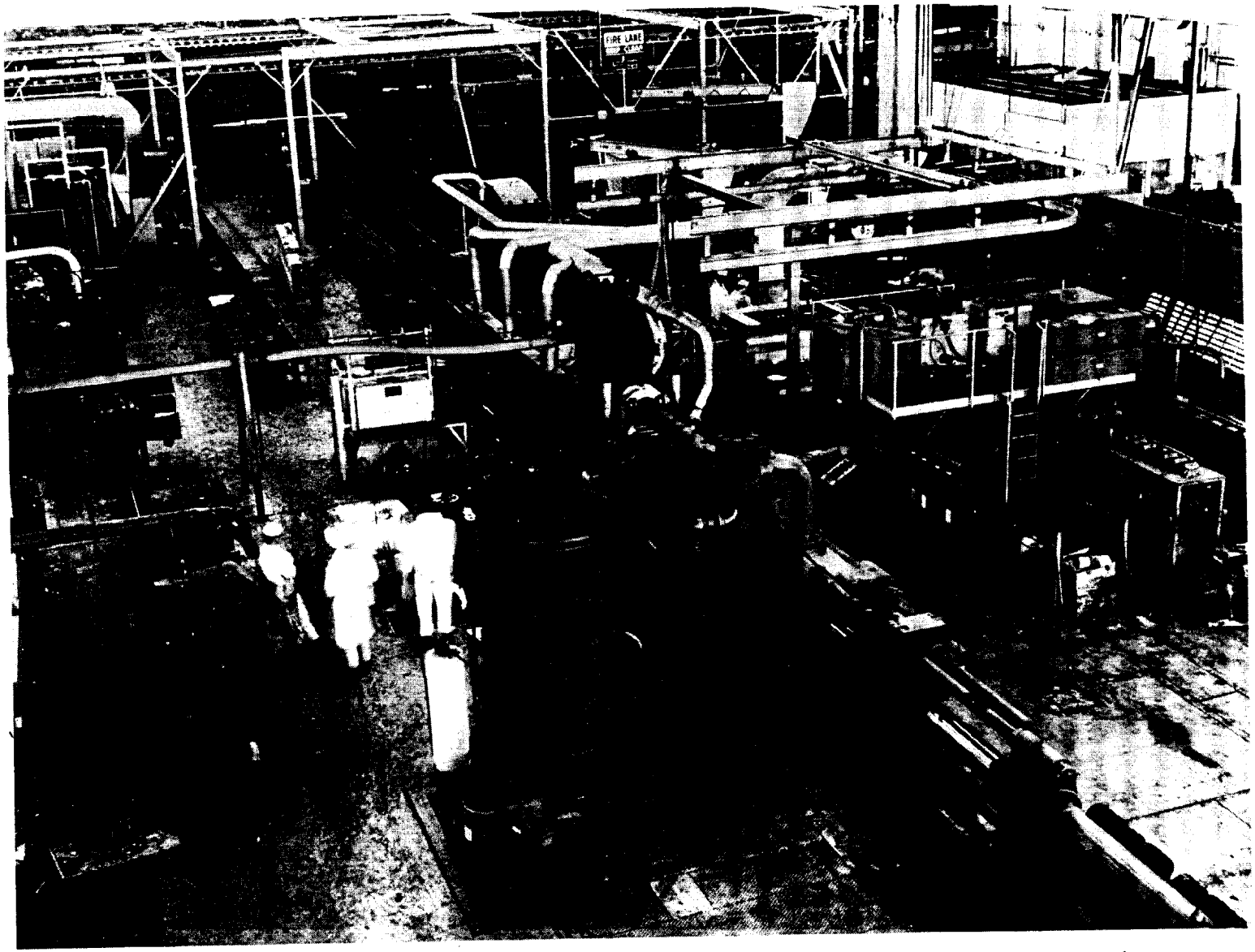

Fig. 2. View of extrusion area showing extrusion press in right foreground.

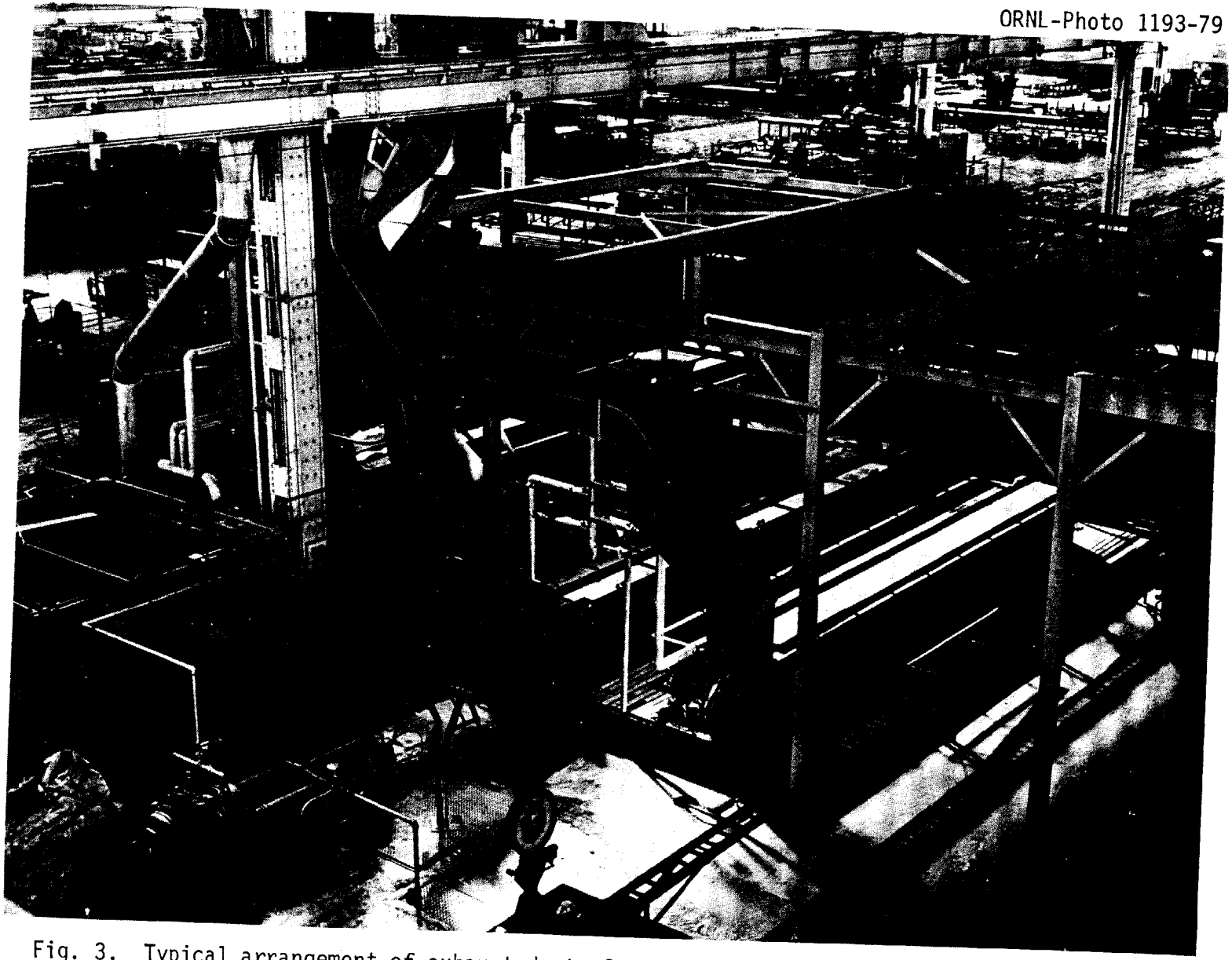

Fig. 3. Typical arrangement of exhaust ducts for cutting, etching, and extrusion operations.

the operations area. Because of some uncertainty regarding the age of this wall, the survey of fixed alpha and beta-gamma activity on the floor was extended to the areas immediately adjacent to and on the other side of the concrete block wall. It was later learned that these walls were erected in early 1960 prior to termination of operations in 1961.

For convenience, a plan view of the shaded area in Fig. 1 is presented in Fig. 4. For the purposes of this survey, a plan view including the shaded area in Fig. 1 as well as the area outside the concrete block walls, which are now removed, is shown in Fig. 5. The grid shown in this figure represents an alpha-numeric code assigned by the original architect. This code is assigned to vertical steel beams which were placed along the outer walls of the facility. The plus signs (+) which appear in these figures represent the intersection of lines of projection drawn between these vertical beams. Because of the massive roof framing design, the need for vertical beams in interior spaces was minimized. Within the operational area (nonsymmetric with respect to the remainder of the building) there were three of the beams at grid locations 17N, 21N, and 25N.

RADIOLOGICAL SURVEY MEASUREMENTS

Most measurements made during the course of a formal survey of former MED/AEC facilities follow a fixed pattern and are designed to permit an evaluation of the current radiological status of the property and to permit comparison with guidelines for the release of decontaminated property for unrestricted access. Differences in measurement protocol between sites are generally attributable to the type of material handled and to the type of operation. The measurements listed in this section, therefore, represent those which were deemed necessary in order to provide an adequate survey of the former Bridgeport Brass plant.

Gamma-Ray Measurements

Gamma-ray exposure-rate measurements were made 1 m above the floor throughout the area used for extruding uranium. The spacing used for these measurements was chosen according to the alpha-numeric grid

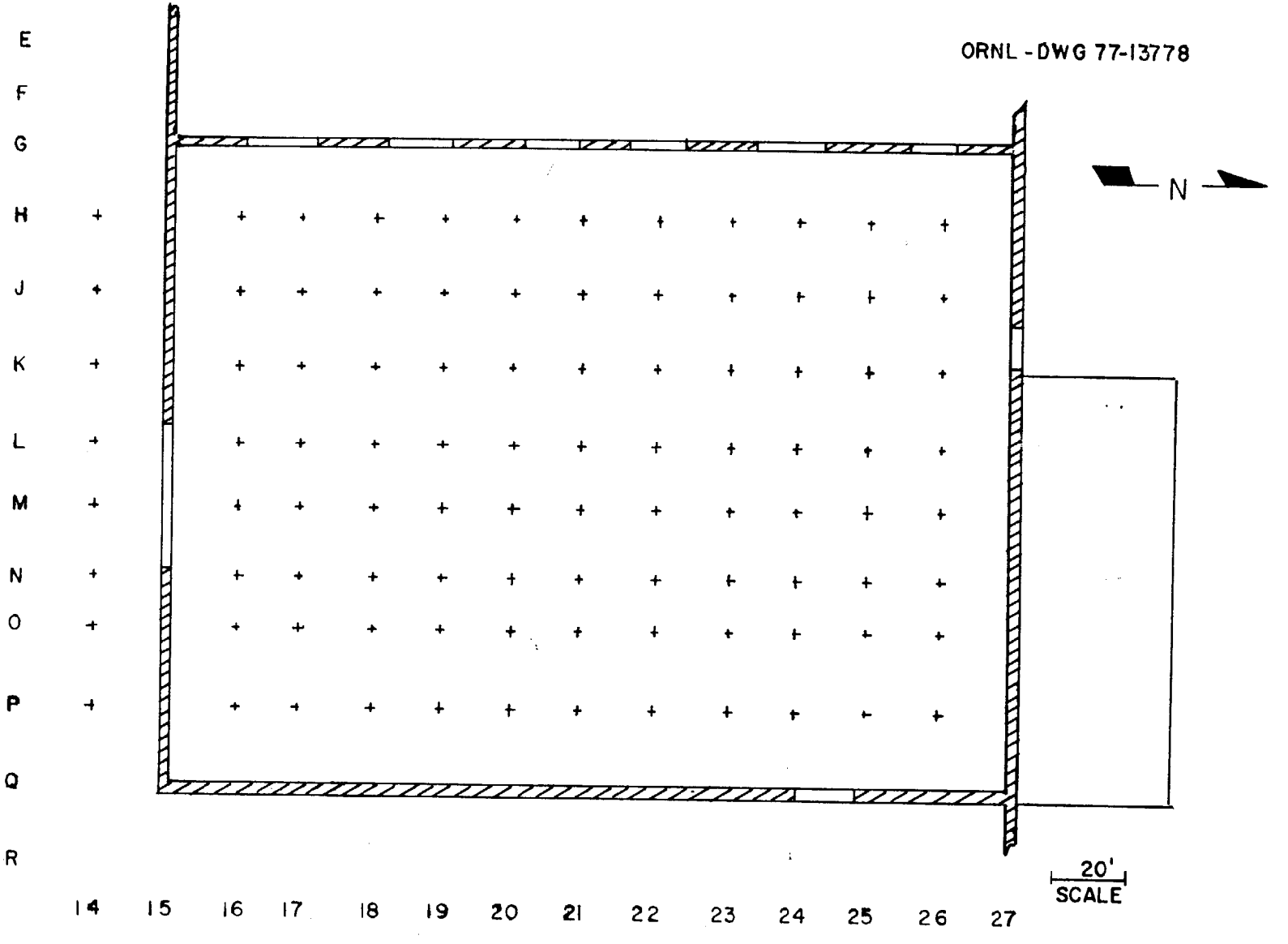

Fig. 4. Plan view of operations area showing building grid spacing.

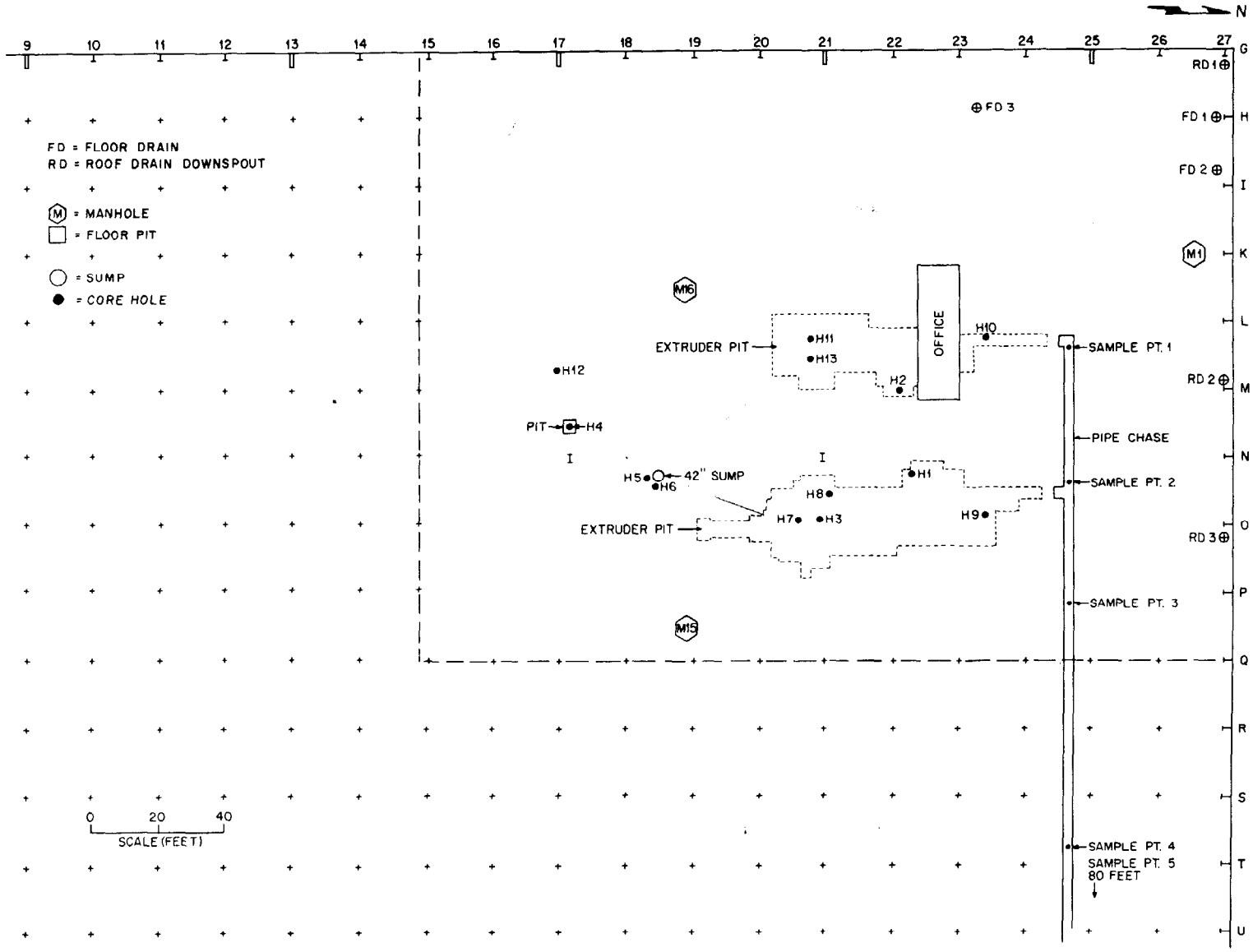

Fig. 5. Plan view of operations area showing location of floor drains.

depicted in Fig. 4. Also, gamma-ray exposure-rate measurements were made 1 m above the ground in the open land area north of the building shown in Fig. 6, as well as in the building, inside manholes, inside down spouts of roof drains, sumps, and the pipe chase at locations indicated in Fig. 5.

Measurements of Alpha and Beta-Gamma Contamination on Building Surfaces

In order to determine the levels of total and transferable uranium contamination on surfaces, the following surfaces were monitored: structural steel, elevated catwalks, pipes, heaters, light fixtures, vertical walls, and other objects above the floor level; floor of the plant area involved in the AEC contract and the floor in areas adjacent to the operational area; and the roof surface including the roof storm drain troughs.

"Contamination," as used in this report, refers to radioactive materials, whether fixed or transferable, deposited in or on surfaces. Survey meter readings made on surfaces were used to indicate the levels of total surface contamination, while standard smear techniques were used to determine the levels of transferable contamination.

Concentration of Radionuclides in Air

Filtered, high-volume air samples (50,000 to 100,000 liters) were collected in three locations in the general area of uranium extrusion. The filters were returned to ORNL and analyzed to determine the amount of ^{234}U, ^{235}U, and ^{238}U from which air concentrations of these nuclides could be calculated.

Investigation of Underground Drains

There is one main storm sewer near the operations area of the main building. The sewer is used as drainage for water collected on the roof. Air from exhaust ducts near the extrusion and cutting area was released at some point above the roof. Local fallout from these ducts was then washed into the storm drain during rainy periods. Three floor

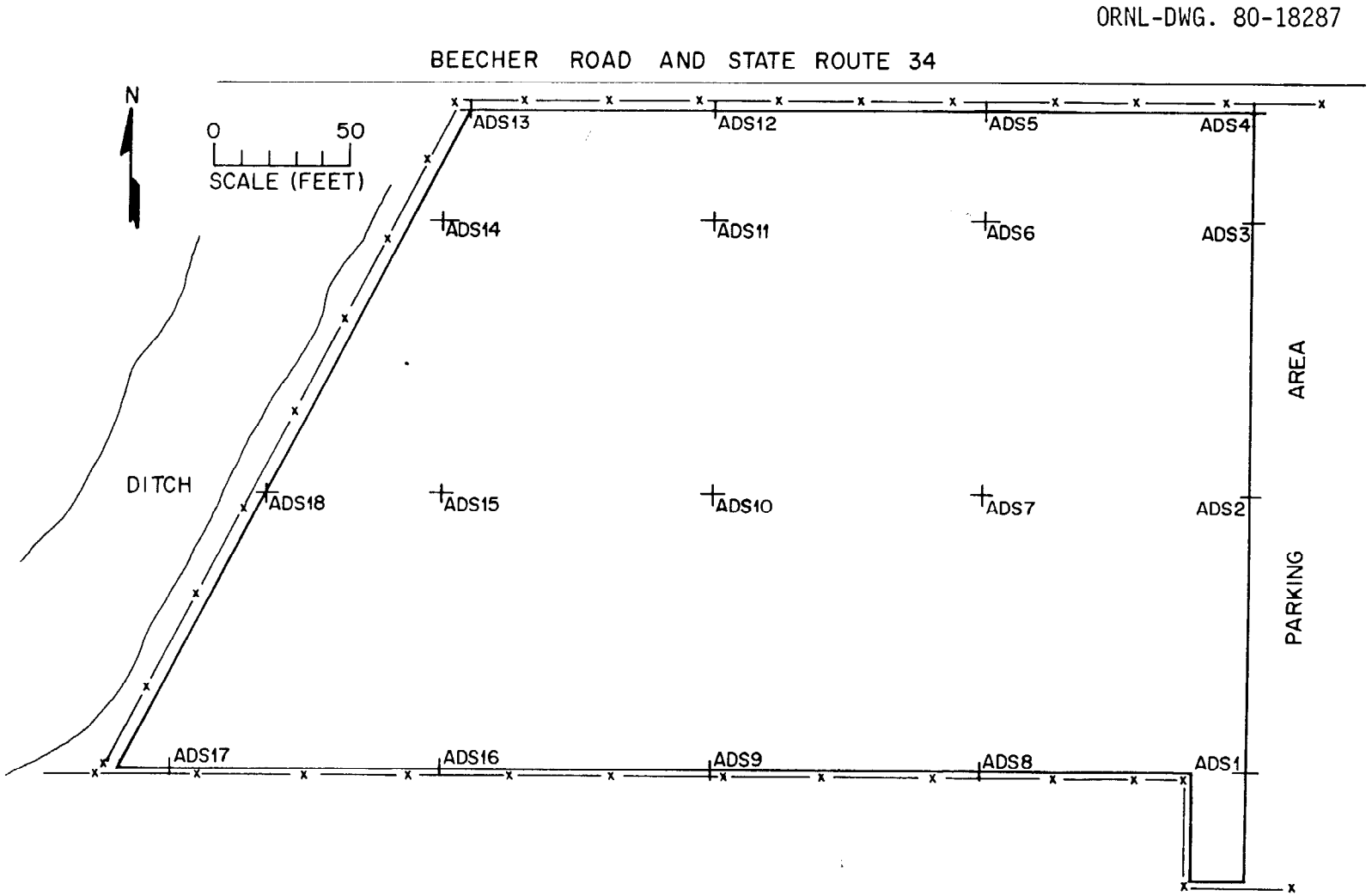

Fig. 6. Plan view of open ground area north of the building.

drains were found in the building. These are shown in Fig. 5. Additionally, service manholes M1, M15, and M16 shown in Fig. 5 contained drains at the bottom which terminated in sump SP1 located on the loading dock at the north end of the building. A 42-in.-diam sump, shown in Fig. 5, contained two drains near the bottom. One entered the sump from an eastward direction and the other from a westward direction. No information is available as to the origin of these drains. Samples from this main trunk line of the storm sewer were collected at three points (see Fig. 1) and analyzed at ORNL. Also, samples from all other underground drains were collected. The concentration of ^{238}U was measured in all samples.

Concentrations of Radionuclides in Soil Collected on the Site and off the Site

Samples of soil were collected on three sides of the main building (see Fig. 1) from the open land area north of the plant (see Fig. 6) and at four places in Adrian from 2 to 5 miles from the Bridgeport Brass plant. These samples were analyzed at ORNL for ^{238}U and for ^{226}Ra.

RESIDUAL CONTAMINATION GUIDELINES

Through an investigation of the operations of this facility, it was determined that only uranium was processed. The U.S. Nuclear Regulatory Commission (NRC) has published guidelines[2] for the decontamination of facilities and equipment prior to release for unrestricted use. Surface alpha radioactivity limits in these guidelines for uranium are 5,000 dpm/100 cm^2 averaged over 1 m^2 and 15,000 dpm/100 cm^2 maximum. For associated beta-gamma contamination, the radiation dose rate limits at 1 cm above surfaces are 0.2 mrad/h averaged over 1 m^2 and 1.0 mrad/h maximum for an area not greater than 100 cm^2. The NRC guidelines are presented in Appendix I. At the time survey operations began at this site, these guidelines had not been adopted for the DOE (then ERDA) resurvey program. Therefore, for the floor of this facility, it is not possible to compare the results of early measurements except in a general way. In December, 1976, the NRC guidelines were adopted for the

program since they were being used by NRC for the release of decontaminated property for unrestricted use. In comparing the results of the survey with the NRC guide, it was obvious that residual contamination in some areas exceeded limits given therein. This information was passed on to the current property owner who, in turn, performed additional decontamination and requested that another survey be made of the facility. This survey was conducted during February and March, 1977. Specific survey procedures used during surveys are presented later. Those of the second survey were designed to permit comparison with the NRC guidelines.

DESCRIPTION OF SURVEY PROCEDURES

Prior to conducting the initial survey, the nature of the extrusion operations was discussed with plant personnel (some of whom worked at the plant during the Bridgeport Brass operation), and an inspection was made of the building. Based on these discussions and observations made during the inspection, it was decided that a representative survey could be accomplished by measuring contamination levels on (1) floor and wall surfaces; (2) elevated chord trusses, heaters, light fixtures, pipes, etc., at the 35-ft level; (3) structural steel at the 45-ft level; (4) catwalks, windows, hand railings, and vertical surfaces in the east and west "monitors" between 45 and 55 ft above the floor; (5) roof, including original exhaust outlets; (6) loading and unloading ramp on the north side of the building; and (7) floor and roof drains. Because some contamination existed on the overhead structure, it was important to obtain samples of air to determine whether survey activities resulted in airborne contamination. For this reason, it was felt that high volume air samples should be collected during the survey.

Readings of surface radioactivity and gamma-ray exposure rate were made in all areas which were accessible. However, it should be stressed that during the initial survey there were some areas of the floor which were covered or otherwise rendered inaccessible because of material stored on pallets and equipment stored on the floor directly. These places which were only partially available for survey are indicated as

shaded areas in Fig. 7. There are a total of 130 grid points within the boundaries of the operational areas. Measurements were made at 44 grid points on interior floor surfaces and at 23 grid points along the four walls. The area covered around each interior grid point was approximately 3.5 m^2 and one-half that area at grid points along the walls (this represents approximately 17% of the floor area actually surveyed). Random alpha and beta-gamma readings were taken in each square meter within the area, and the average of those random readings was recorded for each grid point area. One random smear sample was taken within the area covered at each grid point. In addition, 100 smear samples were taken at random locations on the floor throughout the operational area.

Prior to conducting subsequent survey measurements at this plant, all stored material and equipment were removed, and the floor area was cleared. For this part of the survey, a revised survey plan was prepared and approved by DOE-ORO (then ERDA-ORO). It gave details of the techniques utilized for taking surface contamination readings.

There were three sets of off-gas ducts which had been cut at a height of approximately 15 ft above the floor (the lower portion was scrapped when the plant was decommissioned). The lower end of the duct was covered, and the duct was intact from that point to the ceiling. Sample scrapings were taken from the inside of these ducts and returned to ORNL for analysis.

After learning that service pits existed beneath the extrusion units (see Fig. 8), and that after removal of the equipment the pits were filled with sand and covered over with concrete at original floor level, plans were developed to investigate these areas. This survey was conducted in April, 1979. The plan included (1) direct gamma radiation level measurements at sequential depths (typically 1 ft) in holes drilled through the concrete floor, sand, and into concrete surfaces in the bottom of the service pits (see Fig. 5); (2) analysis of core samples from surfaces of the bottom of service pits; (3) analysis of scale, sediment and crud from surfaces of drains, sumps, service manholes, and the pipe chase (located toward the north end of the building); and (4) sampling of water (or other forms of liquid) from sumps, drains, storm sewer system, etc., where present. Additionally, overhead areas east and south

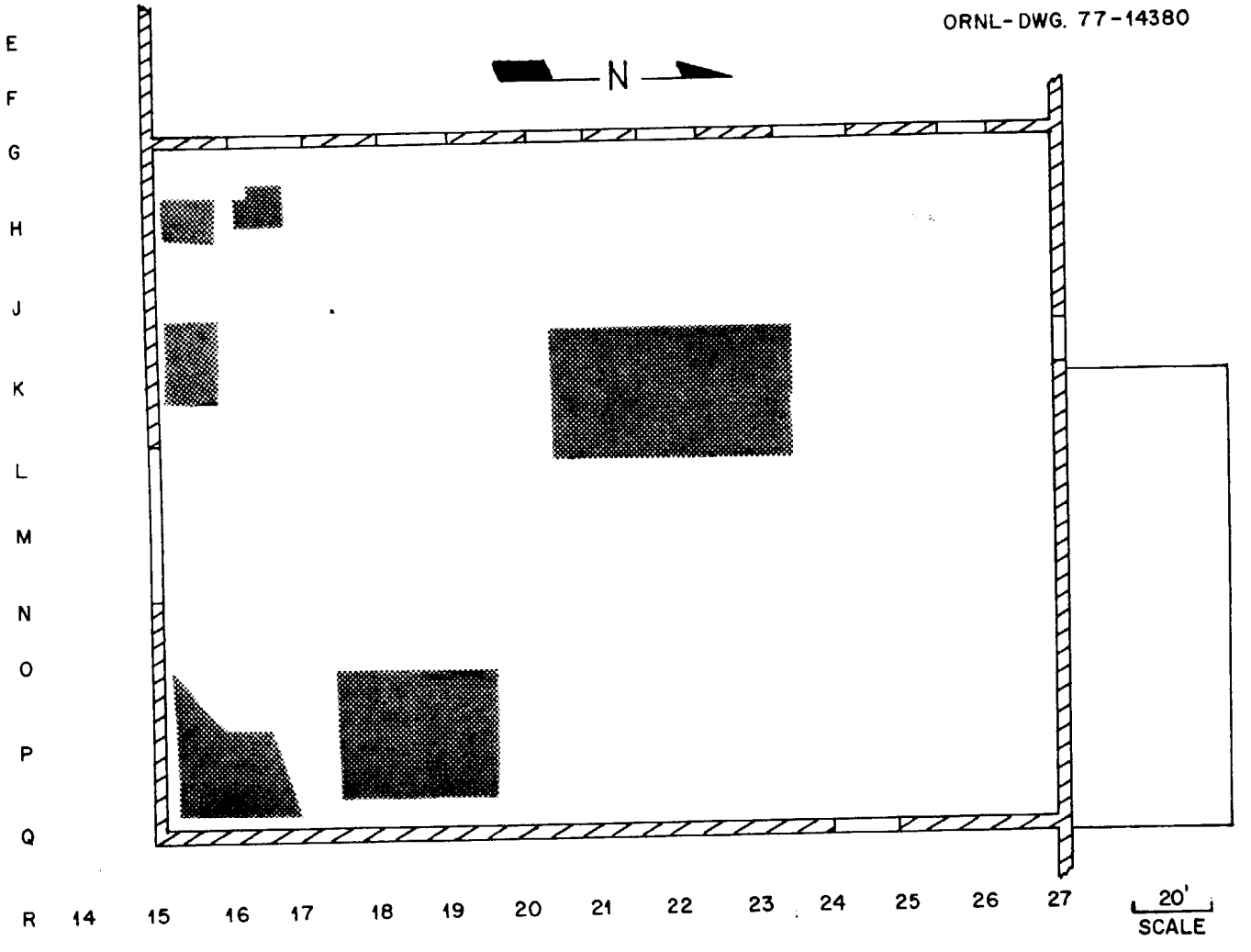

Fig. 7. Inacessible areas of floor during initial survey.

ORNL-Photo 5444-80

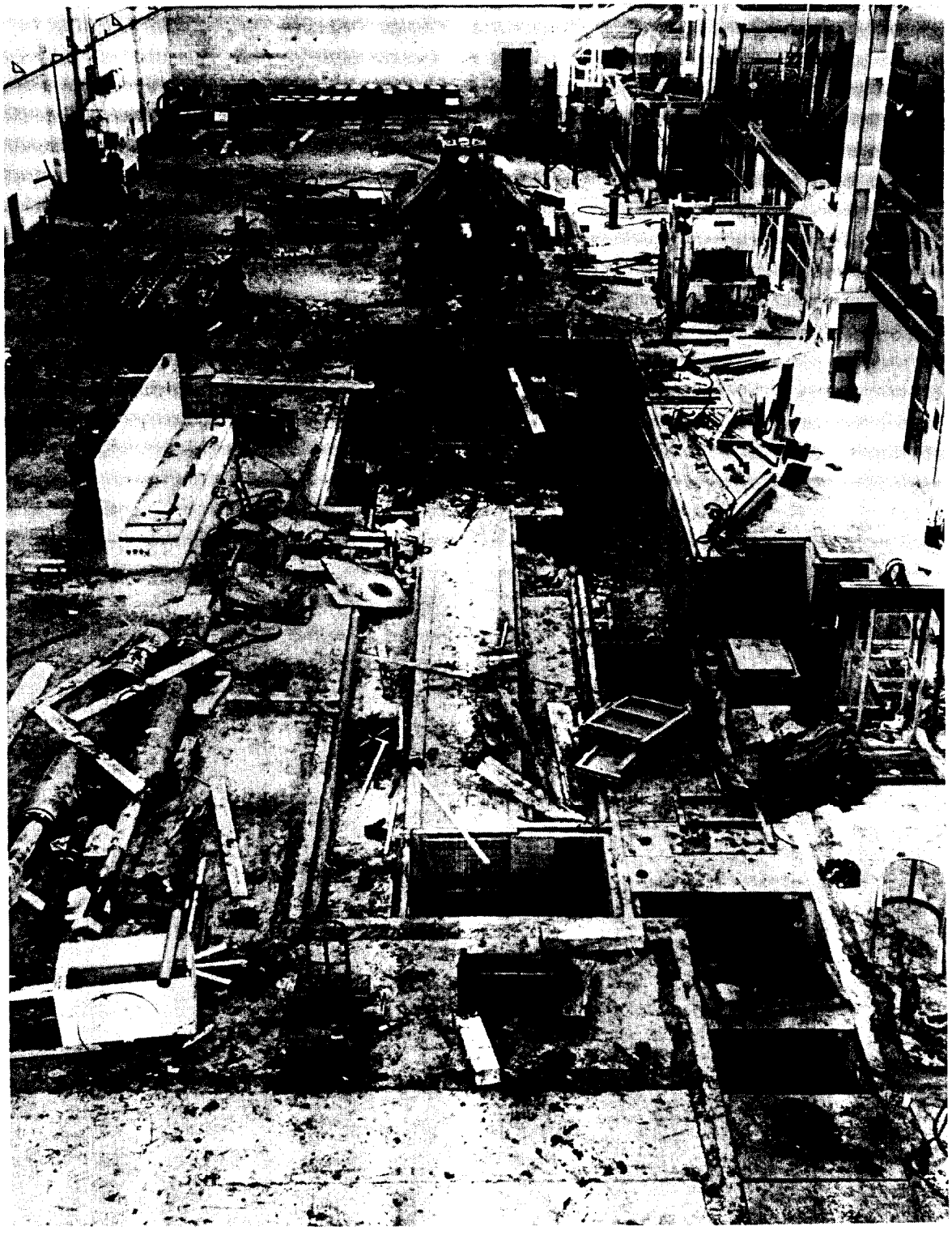

Fig. 8. View looking north to south of extrusion facility during dismantling.

of the operating area, beyond where block walls once existed, were surveyed in a manner as done above the operating area in the "initial" survey.

INSTRUMENTATION AND EQUIPMENT

All measurements of alpha and beta-gamma surface contamination in the operations area were made with portable, hand-held instruments described in Appendix II. Environmental samples which were returned to ORNL were analyzed for radionuclide concentration using state-of-the-art, computerized electronics as described in Appendix III.

Samples of airborne particulate radioactivity were collected with a Staplex, model TF-1A, high-volume sampler. The effective area of aerosol collection was 66.5 cm^2. Whatman No. 41 filter paper was used. This filter is used for fast filtering of coarse and gelatinous precipitates in laboratory work, and it contains 0.01% ash or less. Six air samples collected during the three-day survey period were returned to ORNL for analysis. The filters were ashed, dissolved, and analyzed for uranium isotopes using mass spectrometry techniques.

RESULTS OF RADIOLOGICAL SURVEY

The findings of the surveys presented here show the initial (1976) and current (1979) levels of residual contamination in the facility. All direct meter readings reported here represent gross readings; background radiation levels have not been subtracted. Similarly, background levels have not been subtracted from radionuclide concentrations measured in environmental samples and building materials.

In the initial survey, radioactive residues were found inside the section of off-gas ducts which had not been torn down. Measurements on the inside of these ducts revealed the presence of uranium residue. Because of a buildup of "scale," the direct reading for alpha radioactivity did not exceed 1,200 dpm/100 cm^2; however, the transferable contamination also ranged up to 1,200 dpm/100 cm^2. A direct beta-gamma

reading on a wet paper towel "wipe" revealed up to 8,000 cpm transferable with a corresponding direct beta-gamma reading at the duct surface of 7.4 mrad/h. A sample of scale was taken from the inside of these ducts. Neutron absorption techniques[3] were used to estimate the concentration of uranium in this material. The observed range of ^{238}U was 11,000 pCi/g (3% uranium) to 25,000 pCi/g (7.5% uranium). On the basis of this information, DOE (then ERDA) requested the removal of the ducts. General Motors personnel removed these ducts and cleaned all residue in the area using a filtered vacuum cleaner.

Results of the initial and second transferable surface contamination surveys are presented in Table 1. A given value posted in Table 1 represents net disintegrations per min per 100 cm^2 alpha or beta-gamma activity obtained by use of standard smear techniques applied to grid blocks indicated. Columns I and II refer to results obtained during the first survey. Columns III and IV refer to results obtained during the second survey after General Motors personnel had cleaned some areas. For direct readings of radiation levels on floor surfaces for both the initial and second surveys, results are presented in Table 2. A given value posted in Table 2 represents average direct readings per square meter in an area approximately 3.5 m × 3.5 m. It is seen that the residual β-γ contamination levels on the floor in several areas (see Column II, Table 2) exceed the recommended NRC guidelines for uranium. General Motors personnel removed all material from the operations area and decontaminated those areas where contamination was found in excess of guidelines. It was necessary to remove some sections of concrete by chipping, to remove imbedded steel anchor bolts from the floor, to remove expansion joint packing, and to etch some painted surfaces with acid. All decontamination work was completed in February, 1977. The second floor survey was completed on March 2, 1977. Results of this survey are presented in Columns III and IV of Table 2.

In the second survey, a measurement grid was established for each of the five shaded areas given in Fig. 9. Grid lines were separated by 2 m thus describing a "checkerboard" pattern as shown in Fig. 10. Five random alpha and five random beta-gamma measurements were made in each alternate 1 m^2 area within each gridded area. An area weighted average

Table 1. Results of transferable alpha and beta-gamma contamination measurements of floor during initial survey (Columns I and II) and after cleaning by General Motors (Columns III and IV)

Survey block shown in Fig. 4	I Alpha (dpm/100 cm^2)	II Beta-gamma (dpm/100 cm^2)	III Alpha (dpm/100 cm^2)	IV Beta-gamma (dpm/100 cm^2)
G15	<10	<10	--[a]	--[a]
G16	--[a]	--[a]	<10	<10
G18	--	--	<10	<10
G21	<10	100	--	--
G23	<10	<10	--	--
G25	<10	<10	--	--
G27	<10	10	--	--
H15	--	60	<10	70
H16	<10	50	<10	60
H17	--	--	<10	60
H18	<10	110	--	--
H20	<10	50	--	--
H22	<10	<10	--	--
H24	<10	60	<10	60
H26	<10	40	<10	40
J15	<10	40	--	--
J16	--	--	<10	<10
J17	<10	40	<10	<10
J18	--	--	<10	<10
J19	20	100	--	--
J20	10	10	10	100
J21	<10	110	--	--
J23	<10	30	--	--
J25	<10	<10	--	--
J27	<10	60	--	--
K15	<10	90	--	--
K16	<10	80	<10	<10
K17	--	--	<10	<10
K18	<10	120	<10	<10
K20	--	--	<10	<10
K22	<10	70	--	--
K23	--	--	<10	<10
K24	<10	<10	--	--
K26	<10	<10	--	--
L15	30	<10	<10	<10
L16	--	--	<10	<10
L17	30	<10	<10	<10
L18	--	--	<10	<10
L19	10	80	--	--
L21	20	170	--	--
L22	--	--	<10	<10

Table 1. (continued)

Survey block shown in Fig. 4	I Alpha (dpm/100 cm^2)	II Beta-gamma (dpm/100 cm^2)	III Alpha (dpm/100 cm^2)	IV Beta-gamma (dpm/100 cm^2)
L23	<10	<10	--	--
L25	<10	100	--	--
L27	<10	60	--	--
M15	<10	60	--	--
M16	20	60	<10	<10
M17	--	--	<10	<10
M18	10	40	<10	<10
M19	--	--	<10	<10
M20	20	190	30	70
M21	--	--	50	130
M22	<10	50	50	100
M24	<10	<10	--	--
M26	10	60	--	--
N15	80	150	50	<10
N16	--	--	50	<10
N17	50	60	100	<10
N18	--	--	<10	<10
N19	10	30	50	<10
N20	--	--	50	320
N21	30	50	70	100
N22	--	--	50	160
N23	<10	30	--	--
N25	<10	50	--	--
N27	<10	<10	--	--
O15	10	20	40	300
O16	<10	60	50	<10
O17	--	--	50	400
O18	<10	90	50	400
O19	--	--	<10	<10
O20	70	210	50	<10
O21	--	--	80	<10
O22	10	80	<10	400
O23	--	--	<10	<10
O24	<10	<10	--	--
O26	<10	<10	--	--
P15	<10	30	<10	50
P16	--	--	<10	<10
P17	<10	<10	<10	<10
P18	--	--	<10	<10
P19	20	<10	--	--
P20	--	--	50	100
P21	<10	30	50	<10
P23	<10	<10	--	--
P25	--	--	<10	<10

Table 1. (continued)

Survey block shown in Fig. 4	I Alpha (dpm/100 cm^2)	II Beta-gamma (dpm/100 cm^2)	III Alpha (dpm/100 cm^2)	IV Beta-gamma (dpm/100 cm^2)
P26	<10	10	--	--
P27	<10	<10	--	--
Q15	20	<10	--	--
Q16	10	90	<10	<10
Q17	--	--	<10	<10
Q18	--	--	<10	<10
Q19	--	--	<10	<10
Q20	10	90	--	--
Q21	--	--	<10	<10
Q22	<10	<10	--	--
Q24	<10	10	--	--
Q26	<10	60	--	--

[a]Absence of data value indicates that no measurement was taken in this block during this particular survey.

Table 2. Results of direct alpha and beta-gamma measurements on floor during initial survey (Columns I and II) and after cleaning by General Motors (Columns III and IV)

Survey block shown in Fig. 4	I Alpha (dpm/100 cm^2)	II Beta-gamma (mrad/h at 1 cm)	III Alpha (dpm/100 cm^2)	IV Beta-gamma (mrad/h at 1 cm)
G14 [b]	<50	<0.02	--[a]	--[a]
G15S [b]	<50	<0.02	--	--
G15N [b]	--	0.05	--	--
G16	--	--	<50	<0.02
G18	--	--	<50	<0.02
G21	150	0.02	--	--
G23	<50	<0.02	--	--
G25	<50	<0.02	--	--
G27	<50	<0.02	--	--
H14	<50	<0.02	--	--
H15S	<50	<0.02	--	--
H15N	60	0.03	70	0.02
H16	40	0.02	70	0.02
H17	--	--	70	0.01
H18	50	0.03	--	--
H20	--	--	<50	<0.02
H21	--	--	<50	<0.02
H22	<50	<0.02	--	--
H24	<50	<0.02	--	--
H26	<50	<0.02	--	--
J14	<50	<0.02	--	--
J15S	<60	<0.02	--	--
J15N	140	0.05	--	--
J16	--	--	230	0.04
J17	100	<0.02	<50	<0.02
J18	--	--	<50	<0.01
J19	90	0.01	90	0.01
J20	120	0.03	--	--
J21	70	<0.02	--	--
J23	<50	<0.02	--	--
J25	<50	<0.02	--	--
J27	<50	<0.02	--	--
K14	<50	<0.02	--	--
K15S	<60	<0.02	--	--
K15N	300	0.03	120	0.04
K16	300	0.02	140	0.03
K17	--	--	190	0.03
K18	70	0.10	180	0.02
K19	--	--	<50	<0.02
K20	--	--	<50	<0.02
K21	--	--	<50	<0.02
K22	120	<0.01	--	--

Table 2. (continued)

Survey block shown in Fig. 4	I Alpha (dpm/100 cm^2)	II Beta-gamma (mrad/h at 1 cm)	III Alpha (dpm/100 cm^2)	IV Beta-gamma (mrad/h at 1 cm)
K24	<50	<0.02	--	--
K26	<50	<0.02	--	--
L15S	50	0.10	--	--
L15N	3400	0.70	450	0.03
L16	--	--	220	0.08
L17	590	0.15	250	0.09
L18	--	--	300	0.04
L19	120	<0.01	120	0.01
L21	150	0.07	--	--
L22	--	--	<50	<0.02
L23	180	0.05	--	--
L25	<50	<0.02	--	--
L27	<50	<0.02	--	--
Dock area	All <50 d/m α - 100 cm^2; <0.02 mrad/h β-γ @ 1 cm			
M14	60	0.02	--	--
M15S	<50	0.10	--	--
M15N	480	0.60	320	0.10
M16	150	<0.02	520	0.10
M17	--	--	350	0.13
M18	290	0.50	440	0.10
M19	--	--	220	0.12
M20	100	0.40	130	0.10
M21	--	--	100	0.06
M22	160	<0.02	70	0.07
M23	--	--	180	0.05
M24	60	0.02	--	--
M26	<50	<0.02	--	--
N14	90	0.05	--	--
N15S	330	0.10	--	--
N15N	3400	0.50	430	0.05
N16	--	--	30	0.05
N17	780	0.09	310	0.11
N18	--	--	670	0.06
N19	120	<0.02	240	0.09
N20	--	--	100	0.04
N21	1100	0.90	140	0.08
N22	--	--	90	0.06
N23	<50	<0.02	--	--
N25	<50	<0.02	--	--
N27	<50	<0.02	--	--
014	60	0.05	--	--
015S	130	0.10	--	--
015N	420	0.03	340	0.05

Table 2. (continued)

Survey block shown in Fig. 4	I Alpha (dpm/100 cm^2)	II Beta-gamma (mrad/h at 1 cm)	III Alpha (dpm/100 cm^2)	IV Beta-gamma (mrad/h at 1 cm)
O16	<50	<0.02	110	0.03
O17	--	--	170	0.05
O18	50	<0.02	190	0.04
O19	--	--	60	0.02
O20	1200	0.40	120	0.04
O21	--	--	80	0.02
O22	<50	<0.02	60	0.07
O24	<50	<0.02	--	--
O26	<50	<0.02	--	--
P14	50	<0.01	--	--
P15N	120	0.02	60	<0.02
P16	--	--	60	0.02
P17	<50	<0.02	<50	<0.02
P18	--	--	40	0.02
P19	<50	<0.02	--	--
P20	--	--	70	0.03
P21	<50	<0.02	80	0.07
P22	--	--	120	0.03
P23	--	--	<50	<0.02
P26	<50	<0.02	--	--
P27	<50	<0.02	--	--
Q14	50	<0.01	--	--
Q15S	50	<0.01	--	--
Q15N	590	0.02	90	0.03
Q16	300	<0.02	90	0.03
Q17	--	--	90	0.02
Q18	--	--	110	0.03
Q19	--	--	110	0.03
Q20	310	0.03	--	--
Q21	--	--	<50	<0.02
Q22	190	<0.02	--	--
Q24	240	0.03	--	--
Q26	<50	<0.02	--	--

[a] Absence of data value indicates that no measurement was taken in this block during this particular survey.

[b] The designations 15S and 15N refer to grid line 15, measurements made on floor on south side of block wall and north side of block wall, respectively.

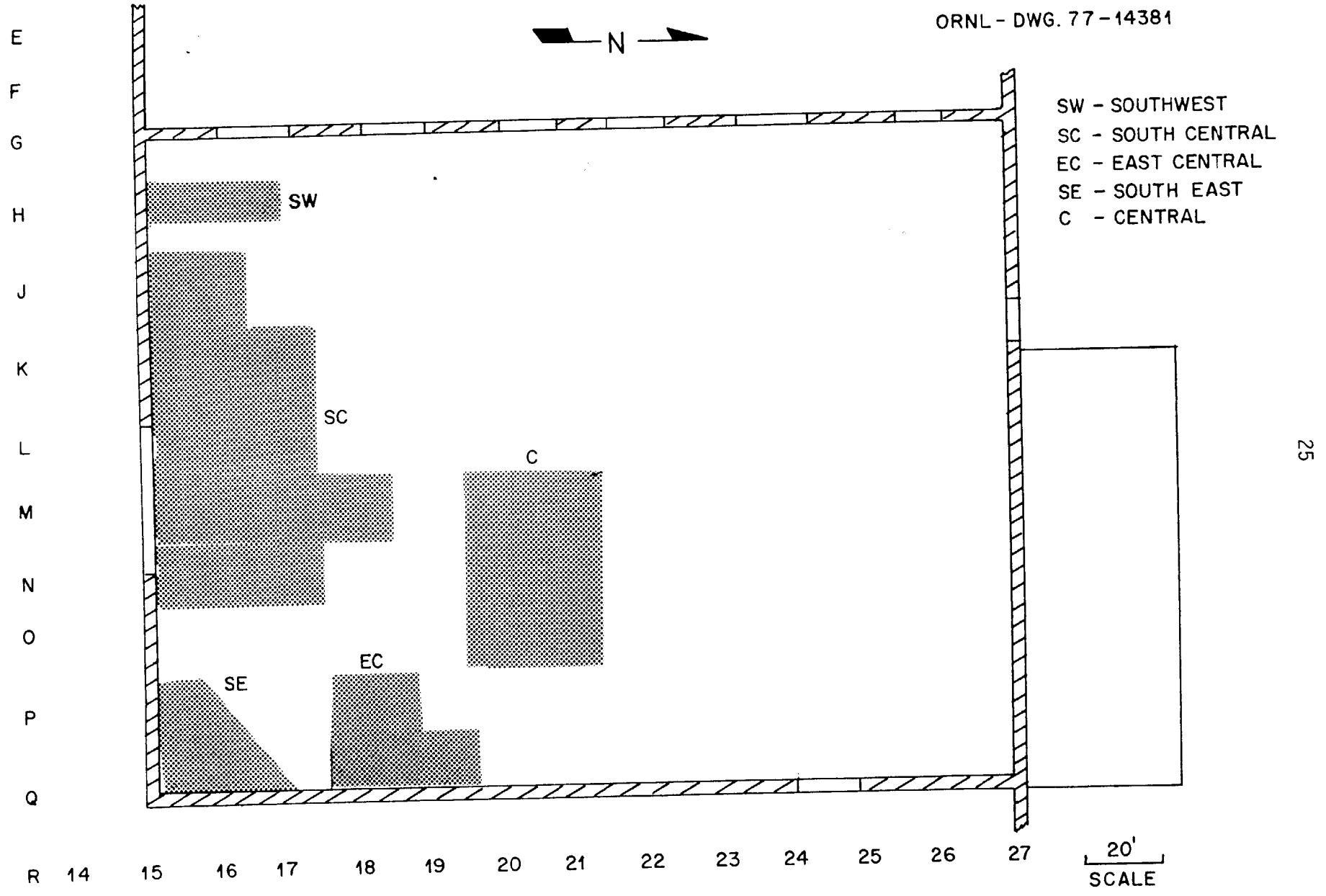

Fig. 9. Sections of building floor included in final survey.

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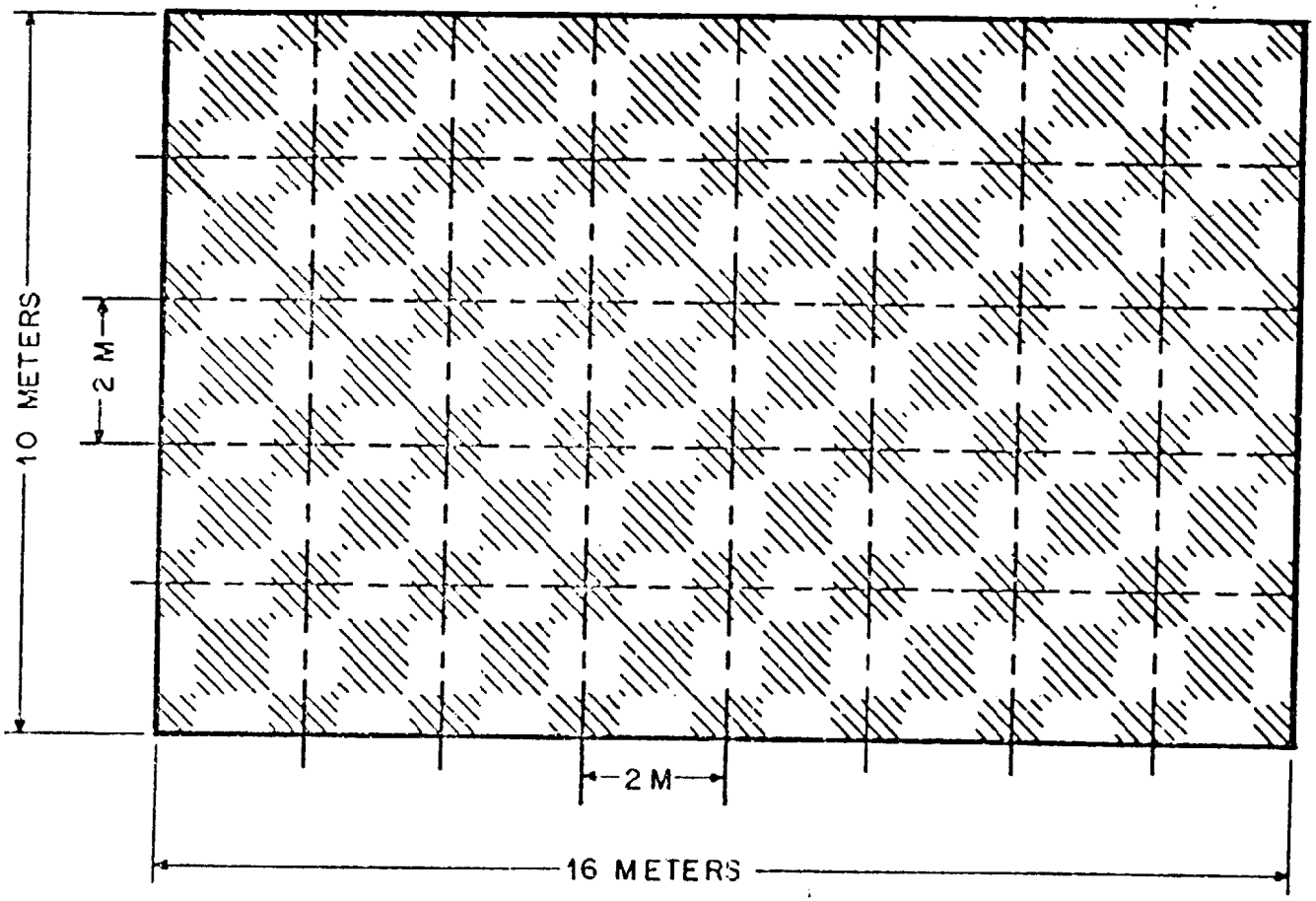

Fig. 10. Example of typical scheme used for making measurement grid.

was determined for these random measurements and recorded as the average reading per 100 cm^2 in each 1-m^2 area actually surveyed. In addition, a smear sample was also taken in each of the 1-m^2 areas in which direct readings were taken. Therefore in the 1977 survey, 50% of the floor area indicated in Fig. 9 was surveyed.

Several 55-gal drums of material were collected by General Motors' personnel during decontamination activities. This material included dust and dirt from vacuum cleaning of overhead beams, trusses, and structures above the floor in June and July, 1976, chipped concrete and expansion joint packing from the floor, and residues collected after removal of building exhaust ducts. These drums were shipped to the DOE Feed Materials Processing Center, Fernald, Ohio, for disposal.

For surfaces above the floor – namely, at 35- and 45-ft elevations – results are given in Tables 3 and 4, respectively. A survey of vertical surfaces (walks, steel beams, pipes, etc.) produced no significant readings above instrument background.

External Gamma-Ray Exposure Rate

The gamma-ray exposure rate at 1 m above the floor in the operations area was measured with portable instruments. The measurements are presented in Fig. 11 and are given in μR/h.

Survey of Storage Shed

A small, wood shed with a concrete floor is located just west of the main building near a rail siding. This building, referred to as the "sheep shed," was used by shipping and receiving for temporary storage of material. A survey of the surfaces in the building did not reveal direct readings of residual contamination in excess of 100 dpm/100 cm^2 for alpha and 0.05 mrad/h at 1 cm above the surface for beta-gamma. Results of smear tests in this area gave no values in excess of 20 dpm/100 cm^2 alpha and 200 dpm/100 cm^2 beta-gamma.

Table 3. Results of direct and transferable alpha and beta-gamma measurements on surfaces at an elevation of 35 ft above the floor

Survey block shown in Figs. 4 and 5	Directly measured contamination		Transferable contamination	
	Alpha (dpm/100 cm^2)	Beta-gamma (mrad/h at 1 cm)	Alpha (dpm/100 cm^2)	Beta-gamma (dpm/100 cm^2)
G10	30	0.02	--	--
G17	60	0.02	<10	<10
G18	40	<0.01	<10	<10
G21	50	<0.01	<10	50
G23	30	<0.01	<10	110
G25	90	0.02	<10	<10
H14	90	0.03	<10	--
H15	180	0.02	--	--
H17	190	0.02	10	<10
H18	80	0.02	20	50
H19	220	0.03	40	30
H21	70	<0.01	10	90
H23	120	<0.01	10	<10
H25	80	<0.01	10	20
J14	90	0.08	5	140
J15	90	0.08	10	--
J17	260	0.02	<10	50
J21	900	0.03	40	40
J25	60	<0.01	10	20
J½17	1500	0.07	120	160
J½19	1400	0.03	30	200
J½21	600	0.02	10	110
J½23	680	0.05	20	170
J½25	130	<0.01	<10	90
K10	50	0.02	<10	--
K14	90	0.08	--	--
K15	70	0.08	<10	40
K15½[a]	240	0.02	20	110
K17	620	0.03	10	80
K19	144	0.05	20	50
K21	130	0.02	<10	30
K23	240	0.02	20	50
K25	300	0.02	<10	180
K½15[a]	400	0.05	20	<10
K½17	450	0.03	10	20
K½19	330	0.05	60	70
K½21	360	0.02	40	80
K½23	240	0.03	10	50
K½25	70	0.02	<10	60
L15	210	0.03	20	140
L17	260	0.03	60	190
L19	300	0.02	20	20
L23	150	0.02	<10	140
L25	70	0.02	<10	20
L½17	240	0.02	20	<10
L½19	150	0.05	10	100
L½21	300	0.02	20	60
L½25	50	<0.01	10	30
M15	320	0.02	40	130
M19	300	0.03	90	40
M21	120	0.03	10	70
M23	50	<0.01	10	50
M25	30	<0.01	<10	<10
N10	50	0.02	--	160
N14	120	0.08	--	--
N15	120	0.03	<10	70
N16	<10	0.02	10	60
N17	200	0.03	10	50

Table 3. (continued)

Survey block shown in Figs. 4 and 5	Directly measured contamination		Transferable contamination	
	Alpha (dpm/100 cm^2)	Beta-gamma (mrad/h at 1 cm)	Alpha (dpm/100 cm^2)	Beta-gamma (dpm/100 cm^2)
N18	30	0.02	10	70
N19	120	0.03	10	100
N20	390	0.07	40	<10
N21	180	0.05	<10	<10
N22	50	0.10	10	70
N23	110	0.02	10	40
N24	60	<0.01	10	<10
N25	160	0.02	<10	<10
O16	300	0.02	20	<10
O18	150	0.02	10	<10
O22	100	<0.01	10	<10
O½19	250	0.02	30	20
O½24	150	0.02	<10	30
P15	70	0.03	--	60
P18	30	0.02	10	50
P19	60	<0.01	<10	10
P22	200	0.02	10	110
P24	50	<0.01	10	70
P26	40	0.01	<10	160
P½16	360	0.02	40	90
P½19	120	<0.01	20	90
P½22	20	<0.01	<10	<10
Q11	90	0.02	--	60
Q16	60	0.02	<10	<10
Q18	80	0.03	10	<10
Q19	50	<0.01	10	30
Q22	70	<0.01	<10	60
Q24	50	<0.01	<10	40
Q26	<20	<0.01	<10	40
R16	160	0.05	--	60
R20	90	0.03	20	20
R24	115	0.03	--	--
R27	--	0.02	80	80
S16	30	0.02	--	--
T15	50	0.03	--	20
T20	90	0.02	--	230
T23	120	0.02	--	40

[a]The alpha-numeric grid used in this table is based on the identification of vertical steel columns as shown on building drawings. Columns running south-to-north in the operational area are labelled 15-27, while columns running west-to-east are labelled G-T. Where a notation such as 15½, or K½, etc. appears in this table, the measurement location is midway between columns 15 and 16, and K and L, respectively.

Table 4. Results of direct and transferable alpha and beta-gamma measurements on surfaces at an elevation of 45 ft above the floor

Survey block shown in Figs. 4 and 5	Directly measured contamination		Transferable contamination	
	Alpha (dpm/100 cm²)	Beta-gamma (mrad/h @1 cm)	Alpha (dpm/100 cm²)	Beta-gamma (dpm/100 cm²)
J15	520	0.14	<10	70
J16½[a]	770	0.14	10	<10
J17½	680	0.17	10	50
J18½	1000	0.17	<10	50
J19	250	0.17	30	60
J19½	580	0.16	10	110
J20	310	0.14	<10	<10
J21	510	0.12	10	<10
J22	380	0.12	<10	90
J23	280	0.07	<10	20
J24	180	0.05	<10	20
J25	70	0.05	<10	<10
J½17	180	0.17	<10	80
J½19	240	0.16	<10	40
J½22	200	0.08	<10	110
J½25	80	0.05	<10	20
K25	100	0.05	<10	10
K½17	210	0.17	<10	<10
K½19	460	0.14	20	90
K½23	70	0.05	<10	70
K½25	100	0.05	<10	<10
L15	350	0.17	20	140
L16½	850	0.14	20	50
L17	820	0.17	10	70
L17½	810	0.17	20	20
L18½	610	0.10	<10	70
L19	640	0.19	60	150
L20	930	0.17	40	20
L20½	630	0.14	10	40
L21½	440	0.10	<10	<10
L22	380	0.10	10	50
L22½	360	0.07	<10	70
L23	210	0.07	<10	70
L24	130	0.05	<10	60
L25	40	0.05	<10	70
O15	480	0.03	<10	20
O16	540	0.10	10	90
O18	330	0.03	<10	110
O19	130	0.14	10	<10
O21	340	0.05	<10	20
O22	420	0.03	<10	<10
O23	330	0.05	<10	<10
O24	200	<0.01	<10	100
O25	200	<0.01	<10	<10
O½17	240	0.03	<10	80
O½19	270	0.03	<10	130
O½20	200	0.05	<10	70
O½21	190	0.04	<10	60
O½22	360	0.04	10	50
O½24	150	0.03	<10	70
O½25	50	<0.01	<10	60
P15	540	0.04	10	70
P16	330	0.03	10	80
P17	270	0.06	10	<10
P18	280	0.03	<10	110
P19	340	0.14	10	80
P20	450	0.03	10	20
P21	350	0.03	<10	20
P22	300	0.03	10	80
P23	450	0.03	<10	50
P24	420	<0.01	10	80
P25	130	<0.01	10	20

[a]The alpha-numeric grid used in this table is based on the identification of vertical steel columns as shown on building drawings. Columns running south-to-north in the operational area are labelled 15-27, while columns running west-to-east are labelled G-T. Where a notation such as 16½, or J½, etc. appears in this table, the measurement location is midway between columns 16 and 17, and J and K, respectively.

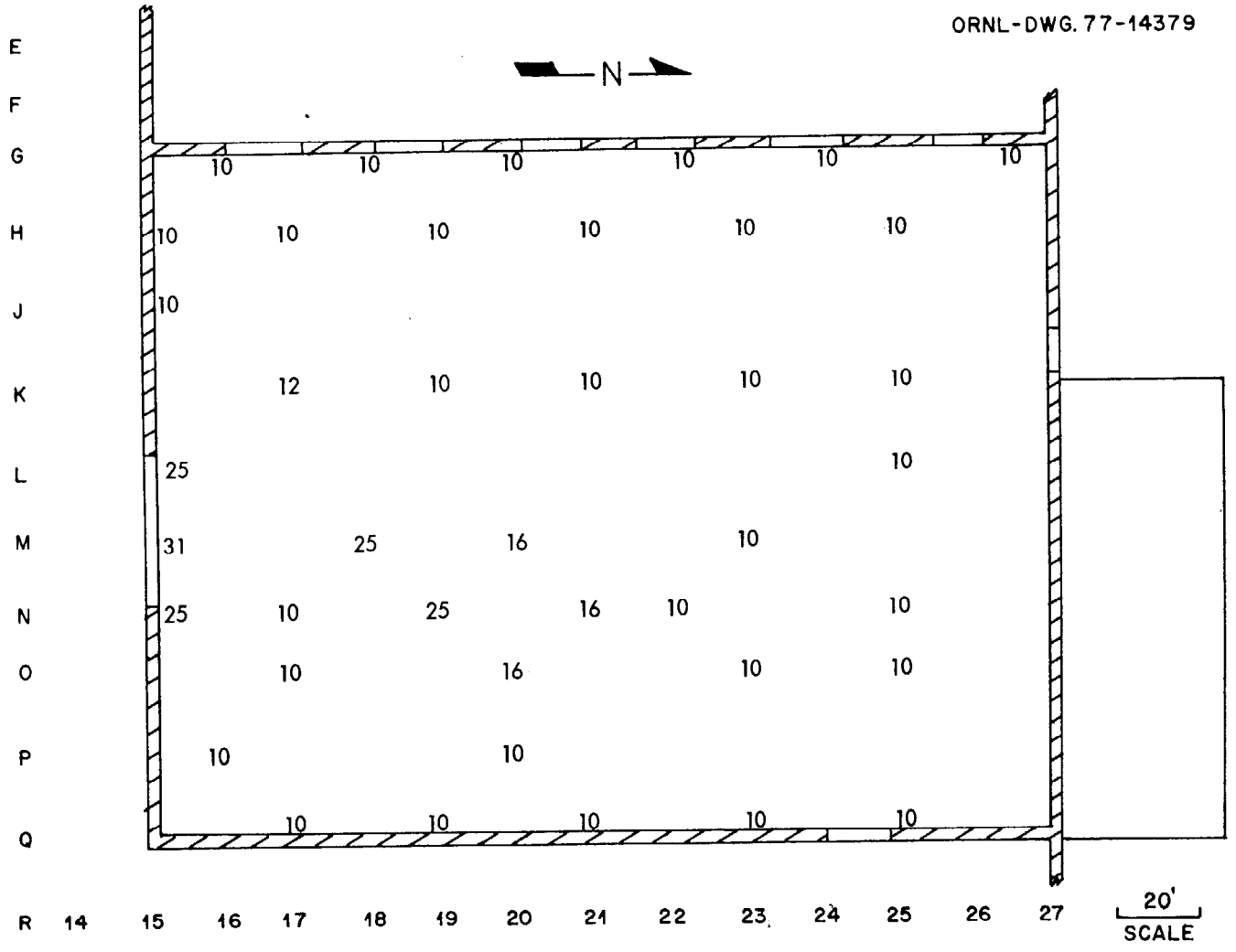

Fig. 11. External gamma-ray exposure rate 1 m above the floor (μR/h).

Storm Drains, Sumps, Manholes, and Pipe Chase

Included in this survey was an investigation of the building storm drain, and a 42-in.-diam sump located inside the building. Only water from the roof drains into the storm system. Sampling points are shown in Fig. 1. Scale which had accumulated throughout the years was sampled at point 1 during the initial survey. Only water samples were collected from the drain at points 2 and 3 in November, 1976, and in February, 1977. Results of the sample analyses revealed uranium concentrations in water ranging from 5 pCi/L to 1800 pCi/L and in solids ranging from 0.1 pCi/g to 1500 pCi/g. Results of these spot samples are presented in Table 5. When the survey of this facility was conducted on March 2, 1977, 900 gal of water was flushed into the drain at point 1 on Fig. 1. One water sample was collected at point 2 (Fig. 2) prior to the arrival of the 900-gal flush, and five water samples were collected at 1-min intervals at this same point as water was flowing through the system. Results of sample analyses are presented in Table 6. Although only minor amounts of uranium contamination were found in these six samples, it may be assumed that, during periods of heavy rains, varying quantities of uranium will be flushed from this drain.

Alpha and beta-gamma direct readings were made on interior surfaces inside the drain access at point 1 in Fig. 1 during the initial survey. No detectable alpha readings above instrument background were noted. However, the beta-gamma reading on accessible drain surfaces was 0.05 mrad/h as measured with a portable Geiger-Mueller (G-M) survey meter. This area was cleaned before the second survey. No readings above instrument background were found at that time.

Water and solid matter were sampled from a 42-in.-diam underground sump located midway between building grids 18 and 19 and midway between N and O. These samples were analyzed for uranium. Results are presented in Table 7.

All samples from both the storm drain and the sump were analyzed for uranium utilizing a neutron-absorption technique whereby delayed neutrons from the fission of ^{235}U in the irradiated samples are counted to determine the concentration of uranium.[3] This technique has a sensitivity of parts per billion (ppb), and it is used routinely to

Table 5. Uranium concentration in water and sediment samples collected in building storm drain manholes

Date of sample	Location of manholes[a]	Type of sample	Concentration of ^{238}U	
			Water (pCi/L)	Solids (pCi/g)
August, 1976	1	Solids	--	1500
November, 1976	1	Solids	--	500
November, 1976	2	Water and precipitate	1800	160
November, 1976	3	Water and precipitate	<5	0.1
February, 1977	1	Water and precipitate	5	700

[a]See Fig. 1.

Table 6. Uranium concentration in samples collected from point 2, Fig. 1 of building roof storm drain as a function of time after dumping 900 gal of water into manhole at Point 1, Fig. 1

Date of sample	Time after flushing	Type sample	Water (pCi/L)	Solids (pCi/g)
March, 1977	0 min	Water and precipitate	0.9	<0.02
March, 1977	1 min	Water and precipitate	2.0	0.02
March, 1977	2 min	Water and precipitate	<1	0.01
March, 1977	3 min	Water and precipitate	5.0	0.02
March, 1977	4 min	Water and precipitate	5.0	0.02
March, 1977	5 min	Water and precipitate	4.0	0.07

Table 7. Uranium concentration in water collected from underground sump tank inside building

Date of sample	Type of sample	Water (pCi/L)	Solids (pCi/g)
August, 1976	Water and precipitate	40	110
February, 1977	Water and precipitate	20	<5
February, 1977	Scale from edge of tank		350

detect normal concentrations or uranium normally found in terrestrial samples.

During the most recent survey of the facility, the 42-in.-diam sump was pumped of contents, and debris was collected for further analysis. This debris consisted of sludge, etc., backed up into two drains which entered the sump from an eastward and a westward direction. Results of analysis of these samples gave 21,000 and 11,000 pCi/g of ^{238}U, respectively.

Direct radiation levels were measured inside all manholes, floor drains, roof drain downspouts at cleanout plugs, inside sumps, and in the pipe chase. Measurements included (1) gamma radiation exposure rate, (2) direct beta-gamma dose-rate measurements at 1 cm from surfaces, and (3) direct alpha radiation measurements. Samples of dirt, crud, scale, water, and oil were collected from these areas and returned to ORNL for analysis.

Results of direct gamma-ray exposure rate, direct beta-gamma dose rate at 1 cm from surfaces, and direct alpha radiation levels for all drains, manholes, sumps, and inside roof drain downspouts are shown in Table 8. Results of the survey made inside the pipe chase are shown in Table 9. Results of analysis of samples of dirt, crud, scale, etc., collected from inside manholes, floor drains, roof drain downspouts, and debris in general, collected elsewhere in the facility, are reported in Table 10. Results of analysis of water and oil samples collected from sumps, manholes, etc., are shown in Table 11.

Core Hole Sampling

After learning that service pits had existed beneath the extruder facility, a decision was made to investigate these areas. In order to do this, it was necessary to drill holes through the existing surface into the former pit areas. Additionally, after finding elevated direct radiation levels inside the bottom of the 42-in.-diam circular sump, a decision was made to drill core holes near it in order to determine if any radioactive material may have seeped from or into this sump. Locations of all core holes drilled are shown in Fig. 5.

Table 8. Results of direct radiation measurements in drains, manholes, sumps, and inside roof drain downspouts

Location	Shown on figure	Drain code	Gamma radiation level (μR/h)		Beta-gamma dose rate at 1 cm (mrad/h)	Direct alpha reading drain (dpm/100 cm²)
			Top of drain	15 to 100 cm below surface		
NW corner, process area	5	FD1[a]	14	13 (2' down)	0.02	<20
15' East of FD1	5	FD2	11	11	0.25	<20
70' South of FD1	5	FD3		30 (0.5' down)	2.0	530[b]
				5 (1.3' down)		
Roof drain	5	RD1[c]	15 (floor of cleanout plug)	11 (average 1' to 4½' down)	0.02	130[d] (inside walls)
Roof (original drain from roof to floor)	5	RD2	8	10 (average 1' 3½' down)	0.05 (inside plug)	45 (inside plug)
Roof drain	5	RD3	8	11 (average 1' to 5' down)	0.01	45 (inside plug)
East slope of loading dock, north side of plant	1	Manhole	27 (halfway to bottom)	23 (bottom)	0.03 (bottom)	--
NE corner of office area	1	Storm sewer manhole		36 (bottom)	0.10 (bottom)	--
North end of building:						
~K27	5	Manhole M1	9	14 (average 1' down to bottom)	0.02	90
~P19	5	Manhole M15	8	20 (average 1' down to bottom)	0.03	320[e]

Table 8. (continued)

Location	Shown on figure	Drain code	Gamma radiation level (μR/h)		Beta-gamma dose rate at 1 cm (mrad/h)	Direct alpha reading drain (dpm/100 cm²)
			Top of drain	15 to 100 cm below surface		
~L19	5	Manhole M16	9	96 (4" from bottom)	5.0	270[f]
				280 (on bottom in water)		
Below floor	5	Circular 42" diam concrete sump		15-150 (1' down to bottom)	1.0 (iron band around tank)	500[g] (maximum on sides of tank)
Inside circular 42" diam sump	5	East drain			2.0 (inside drain)	
		West drain			1.5 (inside drain)	
East slope of loading dock, north side of plant	1	Sump SP1	10	14 (average 1' down to bottom)	--	--
West of office AICA	1	Sump SP2		23 (average 1' to 8' down)	--	--

[a] FD = floor drain.

[b] A smear sample taken from the lip of drain showed 85 dpm transferable alpha contamination and 150 dpm transferable beta contamination.

[c] RD = roof drain.

[d] A smear sample taken from inside walls showed 3 dpm transferable alpha contamination and 80 dpm transferable beta contamination.

[e] A smear sample taken around lip of drain showed 6 dpm transferable alpha contamination and 60 dpm transferable beta contamination.

[f] A smear sample taken around lip of drain showed 90 dpm transferable alpha contamination and 60 dpm transferable beta contamination.

[g] A smear sample taken from sides of tank showed 25 dpm transferable alpha and 190 dpm transferable beta contamination.

Table 9. Results of direct radiation measurements in the pipe chase

Location	Beta-gamma dose rate at 1 cm (mrad/h)	Direct alpha contamination level (dpm/100 cm^2)		External gamma at 1 m above bottom of floor of chase (μR/h)	
		Average	Maximum	Average	Maximum
Sample point 1, far west end	0.13	100	800	--[a]	36
Sample point 2, 40' east of west end of pipe chase	0.23	180	700	36	36
Sample point 3, 75' east of west end of pipe chase	0.02	180	350	--	14
Sample point 4, 155' east of west end of pipe chase	0.02	70	180	13	13
Sample point 5, 235' east of west end of pit	0.02	<20	90	--	13

[a]Absence of data value indicates that no measurement was taken in this block during this particular survey.

Table 10. Results of analysis of dirt, crud, scale, etc., collected from manholes, floor drains, pipe chase, roof downspouts, etc.

Sample	Survey location shown in Fig. 5	^{238}U concentration (pCi/g)
1	Lip of M16	5,600
2	Bottom, M16	15,000
3	Bottom, M1	7
4	FD1	20
5	FD3	11,000
6	M15	260
7	East drain, entering bottom of 42" sump	21,000
8	West drain, entering bottom of 42" sump	11,000
9	Bottom, 42" sump	70
10	Overhead beam at N15	440
11	Crud, lip FD3	8,100
12	Storm sewer manhole, at NE corner office area (see Fig. 1)	210
13	Lip, FD2	480
14	Sample point 1, pipe chase	710
15	Sample point 2, pipe chase	260
16	Sample point 2, top of cross members	510
17	Sample point 3, pipe chase	50
18	Sample point 4, pipe chase	6
19	Sample point 5, pipe chase	0.8

Table 11. Results of analyses of oil and water samples from sumps and manholes

Sample	Survey location (see Fig. 5)	^{238}U concentration (pCi/L)
1	Bottom, 42" diam sump	30
2	Oil on top of water in 42" diam sump (before pumped)	170
3	Oil and water from 42" diam sump (before pumped)	770
4	Water from 42" diam sump after 935 gal pumped out	50
5	Water from 42" diam sump after 1,375 gal pumped out	580
6	Oil from east drain to 42" diam sump (after pumped)	4,100
7	Water from manhole 16	640
8	Oil from on top water in sump 3, outside at north side of building (see Fig. 1)	9,700
9	Water from sump 3	6
10	Water from storm drain manhole, east slope of loading dock, north side building	<0.1
11	Water from storm drain manhole, NE corner office area	1.0
12	Water from core hole 5, inside plant	10

Holes were drilled with a motorized rig equipped with a 7-in.-diam auger, and ranged in depth from 4 ft to a maximum of 13-1/2 ft. The depth chosen was dependent upon location with respect to old service pits and where bottom concrete pit floors were found. A plastic pipe with a 4-in. outer diam was placed in each hole, and a NaI scintillation probe was lowered inside the pipe. Measurements were made both with an unshielded and a shielded probe. The shielded probe was encased in a lead shield with narrow slits on the side. This arrangement allowed measurements of gamma radiation intensities resulting from contamination within small fractions of the hole depth. Measurements were usually made at 1-ft intervals.

Random soil and debris samples were taken from each hole and analyzed for uranium. In addition, some samples from specific depths were collected and analyzed. Results of the analyses of these samples are shown in Table 12. As discussed in Appendix IV, these samples were used to estimate the distribution of ^{238}U contamination in the service pits based on the gamma-ray logging of augered holes. Results of these estimates are given in Table 13. It should be borne in mind that the sources of error discussed in Appendix IV limit the interpretation of this table to providing an indication of the location and relative magnitude of ^{238}U contamination present in the service pits.

Airborne Radioactivity

High-volume air samples were taken during the initial survey period. Samples were normally taken for 2 to 3 hours in the morning and afternoon. A Staplex model TF-1A sampler was used with Whatman No. 4 filter paper. Details of the sampling periods, location, and analysis for total uranium are presented in Table 14. In the analysis of sample 1, it was seen that the ^{235}U concentration (atom percent) was 12.96. This high value was attributed to the small quantity of uranium actually sampled. The concentration of ^{238}U is given in Table 15. The concentration guide (CG) for uranium in air is 3×10^{-12} μCi/ml for continuous nonoccupational exposure. It is seen, therefore, that during the period of sampling, the concentration of uranium in air ranged from a low of

Table 12. Results of analy of soil samples from core holes

Location shown in Fig. 9[a]	Description of sample collection area or depth collected	^{238}U concentration (pCi/g)
H1	Random	2.0
H2	Randon	15
H3	Random	0.9
	0-2" (concrete)	0.5
H4	Random	6.6
	7"-13"	3.9
	13"-19"	2.0
	19"-31"	2.7
	31"-43"	11
	43"-54"	21
H5	Random	1.0
H6	Random	0.9
H7	Random	9.1
H8	Random	1.2
H9	Random	0.7
H10	Random	0.6
H11	Random	100
	Bottom	210
	Concrete	0.4
H12	7"-13"	27
	13"-19"	7.2
	19"-31"	5.4
	31"-43"	24
	43"-55"	20
	55"-63"	1.0
H13	Random	1.5
	Random	1.7

[a] Sample code numbers are identical to core hole numbers.

16.1

Table 13. Estimates of ^{238}U concentration in material in service pits based on augered hole gamma-ray loggings

Location shown in Fig. 9	Depth (ft)	^{238}U (pCi/g)[a]
H1	0	<1
	1	5
	2	7
	3	7
	4	7
	5	4
	6	3
	7	3
H2	0	<1
	1	2
	2	2
	3	2
	4	2
	5	2
	6	4
	6.5	5
H3	0	<1.
	1	6
	2	6
	3	5
	4	4
	5	4
	6	4
	7	3
	8	3
	9	2
	10	2
	11	2
	12	3
H4	0	2
	1	9
	2	10
	3	9
	4	7
	5	7
H5	0	3
	1	5
	2	8
	3	9
	4	7
	5	9

Table 13. (continued)

Location shown in Fig. 9	Depth (ft)	^{238}U (pCi/g)[a]
H5 (cont'd)	6	8
	7	8
	8	8
	9	8
	10	9
	11	11
	12	10
	13	7
	13.5	7
H6	0	4
	1	4
	2	7
	3	8
	4	8
	5	9
	6	9
	7	9
	8	9
	9	7
	10	9
	11	10
	12	9
	12.5	10
	13	7
	14	1
	15	1
	16	1
	17	1
	18	1
H7	0	1
	1	7
	2	6
	3	6
H8	0	1
	1	9
	2	11
	3	12
	4	11
	5	9
H9	0	<1
	1	5
	2	5

Table 13. (continued)

Location shown in Fig. 9	Depth (ft)	^{238}U (pCi/g)[a]
H9 (cont'd)	3	5
	4	5
	4.5	3
H10	0	1
	1	2
	2	4
	3	4
	4	4
	4.5	3
	5	3
H11	0	1
	1	2
	2	4
	3	4
	4	6
	5	9
	6	9
	7	10
	8	20
	8.5	30
H12	0	3
	1	11
	2	10
	3	13
	4	14
	5	12
	6	11
	7	12
	8	12
	8.5	12
H13	0	<1
	1	1
	2	2
	3	2
	4	2
	4.5	2
	5	4

[a] Estimates are based on interpretation of hole loggings described in Appendix IV. Error of these estimates is about 200% for this technique.

Table 14. Total uranium concentration in high volume air samples

Sample	Date	Time		Total flow (liters)	Uranium analysis atomic percent			Total U (μg)	Grid location
		Start	Stop		^{234}U	^{235}U	^{238}U		
1	8/17/76	10:55	13:55	7.1×10^4	0.021	12.96	87.00	0.60	19 X N
2	8/17/76	13:57	17:13	7.8×10^4	0.005	0.722	99.27	14.9	19 X N
3	8/18/76	10:00	13:05	7.3×10^4	0.015	0.860	99.12	0.43	19 X H
4	8/18/76	13:15	17:07	9.2×10^4	0.007	0.723	99.27	277.0[a]	19 X H
5	8/19/76	8:20	10:34	5.3×10^4	0.019	0.792	99.19	0.39	19 X G
6	8/19/76	10:34	12:45	5.20×10^4	0.014	0.721	99.26	1.39	19 X G

[a]One may conclude that physical activity due to the conduct of the survey by personnel and mobile equipment (i.e., moving vehicle used for surveying overhead structures) as the day progressed could have resulted in the higher air activity levels for later periods during the day as material may have become airborne.

Table 15. ^{238}U concentration in high volume air samples

Sample	Grid location	^{238}U (μCi)	μCi/ml
1	19 x N	1.7×10^{-7}	2×10^{-15}
2	19 x N	4.9×10^{-6}	6×10^{-14}
3	19 x H	1.4×10^{-7}	2×10^{-15}
4	19 x H	9.2×10^{-5}	1×10^{-12}
5	19 x G	1.3×10^{-7}	2×10^{-15}
6	19 x G	4.6×10^{-7}	9×10^{-15}
		RCG Air =	3×10^{-12}

0.06% to a high of 33.2% of the concentration guide. Because of the short-term nature of these samples, it is not possible to determine an annual average concentration for airborne uranium.

Alpha and Beta-Gamma Readings on the Roof

Measurements were made on the roof of this facility. A scan of the flat roof surfaces and drain troughs with a G-M survey meter and random measurements with a portable alpha scintillation detector did not reveal readings which were higher than the instrument background readings. There were two areas where off-gas ducts, blowers, and tie-down anchors had been mounted during operational periods. This equipment was removed at the time the plant was decommissioned. Readings on exterior surfaces of the covers of feed-through sleeves for the ducts averaged less than 100 dpm/100 cm^2 alpha, and beta-gamma readings were less than 0.02 mrad/h. Interior surfaces of these feed-through sleeves were also monitored. Readings here averaged 300 dpm/100 cm^2 alpha and 0.05 mrad/h beta-gamma.

On-Site Soil Samples

A random sample of surface soil was collected on each of three sides of the plant building (see Fig. 1) during the first survey and returned to ORNL for analysis. Each sample was dried overnight at 110°C in order to remove moisture, pulverized to a particle size of 500 μm, packaged in plastic counting vials, and stored for equilibration of radon daughter products. The samples were then analyzed using gamma-spectroscopy techniques.[4] Radionuclides which were sought specifically included ^{226}Ra and ^{232}Th. Analysis for ^{238}U was done as described earlier. The results of these samples are given in Table 16.

A survey of the open land area north of the building was conducted (see Fig. 6). This survey included (1) gamma-ray exposure-rate measurements made 1 m above the ground, (2) beta-gamma dose-rate measurements at 1 cm from the ground surface, and (3) collection of a sample of soil at each grid point shown. Results of this survey are shown in Table 17.

Table 16. Analysis of on-site soil samples

Sample[a]	Nuclide concentration in (pCi/g)		
	^{226}Ra	^{232}Th	^{238}U
Soil ADS19	0.57 ± 0.05	0.42 ± 0.04	3.4 ± 0.6
Soil ADS20	0.65 ± 0.05	0.51 ± 0.06	6.9 ± 1.4
Soil ADS21	0.98 ± 0.03	0.62 ± 0.07	1.8 ± 0.8

[a]See location in Fig. 1.

Table 17. Results of measurements in open land area north of building

Surface sample[a] location	External gamma radiation level at 1 m (μR/h)	Beta-gamma dose rate at 1 cm from surface (mrad/h)	^{238}U concentration (pCi/g)
ADS1	11	0.02	1.4
ADS2	13	0.02	1.9
ADS3	12	0.02	1.7
ADS4	11	<0.01	1.5
ADS5	12	0.02	1.4
ADS6	11	0.02	1.0
ADS7	15	0.02	2.0
ADS8	12	0.03	1.1
ADS9	12	0.01	1.7
ADS10	13	0.01	0.7
ADS11	14	0.02	0.9
ADS12	13	0.02	1.0
ADS13	11	0.02	1.1
ADS14	12	0.02	1.0
ADS15	12	0.02	1.0
ADS16	12	0.02	1.0
ADS17	11	0.02	0.9
ADS18	12	0.02	1.0

[a] See Fig. 6.

Off-Site Background Soil Samples

Four samples of soil were collected from the surface at points up to several miles from the Bridgeport Brass facility. These samples were prepared in the same way as those obtained on site, and they were analyzed using gamma scintillation and neutron absorption techniques. Results are presented in Table 18. A comparison of the information in Tables 16, 17, and 18 suggests that some residual uranium may be found on the surface of the ground around the immediate vicinity of the main building. Although the uranium content of on-site samples was slightly higher than that observed in samples from off the site, the uranium concentration is within the normal terrestrial range for large portions of the United States.

SIGNIFICANCE OF FINDINGS

At the request of DOE (then ERDA), a comprehensive radiological survey was conducted in August, 1976, in the operational areas of the former Bridgeport Brass Company Special Metals Extrusion Plant in Adrian, Michigan.

Included in this survey were measurements of residual uranium contamination on building surfaces, external gamma-ray exposure rates in the operations area 1 m above the floor, high-volume air samples in the operations area, uranium in water and residues from underground tanks and drains, and the analysis of uranium, radium, and thorium in samples of soil from on-site and off-site locations.

During the initial survey of the facility, residual contamination in excess of NRC guidelines was found on several sections of the floor. Equipment which had been stored in this area was removed, and contaminated areas were cleaned by General Motors. Also, during the initial survey, several ducts used for exhaust in the extrusion and cutting operations were found to be contaminated with uranium (up to 7.5% uranium in scale scrapings from inside these ducts). These ducts were subsequently removed by General Motors and sent to Fernald, Ohio, for disposal along with other contaminated material collected during General Motors' early cleanup. When a subsequent survey was performed in March,

Table 18. Analysis of off-site background soil samples collected in the vicinity of Adrian, Michigan

Sample location	Nuclide concentration in (pCi/g)		
	^{226}Ra	^{232}Th	^{238}U
Junction of Wellsville and Deerfield Roads	2.0	0.4	0.9
Junction of Bent Oak and Shepherd Roads	1.5	0.8	1.1
Junction of Forrester and Townline Roads	1.5	0.7	1.1
Junction of Gorman and Baker Roads	1.2	0.5	0.7

1977, all areas of the floor and overhead structural members were found to be within the guidelines used for the release of decontaminated facilities for unrestricted use. However, some areas below the floor level were found to be contaminated with uranium.

The concentration of uranium in water and residue removed from a 42-in.-diam. circular underground sump ranged from 20 to 40 pCi/L in water and from 110 to 350 pCi/g in residue and scale collected from upper horizontal edges around the inside top of this tank. Most of the scale which had accumulated in this area was included in the sample. When the most recent survey of the facility was conducted in April, 1979, further investigation of this sump produced an oily sample from an east drain near the bottom which gave 4100 pCi/L of ^{238}U. This same drain contained residue and scale which contained 21,000 pCi/g of ^{238}U. Material from the sump's west drain contained 11,000 pCi/g. Material collected from the bottom of manhole M16 (see Fig. 5) gave 15,000 pCi/g of ^{238}U. Liquid from a sump located on the north loading dock area contained 9700 pCi/L. Samples collected from an underground storm drain contained from 5 to 1800 pCi/L in water and from 0.1 to 1500 pCi/g in solid residue. During the survey, 900 gal of water were flushed down the storm system to determine if significant quantities of material were removed from the drain during heavy rains. Results of analyses on samples collected during the flushing revealed only minor concentrations of uranium in the water. Material from core holes drilled into areas under the floor formerly serving as extruder service pits contained up to 210 pCi uranium per gram of debris from the former concrete floor surfaces.

Air samples collected in the former extrusion area were analyzed for uranium. Although the sampling period was too short to establish an annual average concentration, the airborne uranium concentration ranged from 0.06 to 33% of the concentration guide for continuous nonoccupational exposure.

Samples of soil were collected on the surface near the main plant building and at four locations off the site ranging up to 5 miles distance. A comparison of the concentration of uranium in these samples suggests that minute quantities of residual uranium may exist on the

surface near the plant due to deposition of material exhausted from blowers during extrusion and cutting operations. However, the on-site uranium concentration remains small and is within the normal terrestrial distribution of uranium for large portions of the United States.

An evaluation has been made of current radiation exposures at the former Bridgeport Brass site and is presented in Appendix V 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 Michigan, as well as to scientifically based guideline values established for the protection of radiation workers and members of the general public.

REFERENCES

1. R. D. Heiser, personal communication, November, 1976.
2. *Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Material*, U.S. Nuclear Regulatory Commission, Division of Fuel Cycle and Material Safety, November, 1976.
3. 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).
4. G. D. Kerr, P. T. Perdue, and J. H. Thorngate, "A Ge(Li) Detector System for Laboratory Counting of Natural Radioactivity in Environmental Samples," *Proc. Symposium on Natural Radioactivity in Man's Environment*, Saratoga Springs, New York, October 11-13, 1976.

APPENDIX I

PERTINENT RADIOLOGICAL REGULATIONS, STANDARDS, AND GUIDELINES

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

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 I-1 specify the radioactivity and radiation exposure rate limits which should be used in accomplishing the decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table I-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 I-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:

Table I-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 others noted above.	5,000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1,000 dpm $\beta\gamma$/100 cm^2

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

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

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

[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.

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 I-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.

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

---

[a]Note on application of Tables I-2 and I-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 $1/n \sum_{\eta} Si \geq L$, 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 $1/n \sum_{\eta} Si \geq AL$, 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.

APPENDIX II

DESCRIPTION OF RADIATION SURVEY METERS AND SMEAR COUNTERS

RADIATION SURVEY METERS

Alpha Survey Meters

Two types of alpha survey meters are used to measure alpha radioactivity on surfaces. One type of instrument uses a ZnS scintillator and the other uses a gas-flow proportional counter 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. II-1). The ZnS detector is covered with a 0.28-mil aluminized mylar sheet in order to make the instrument light-tight. The mylar, in turn, is covered with a grid to prevent puncturing the detector when surveying over 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.

The gas-flow proportional counter uses propane gas as the detection medium. Through front panel meter readings, it can be used to measure alpha contamination levels from a few hundred dpm/100 cm^2 to several hundred thousand dpm/100 cm^2. If individual pulses are counted, this instrument can also be used for measurements down to a few dpm/100 cm^2. The probe has a surface area of approximately 61 cm^2 and has a 0.25-mil aluminized mylar covering with a protective grid. Due to the protective grid, the active area of the probe is 50 cm^2. It is relatively insensitive to other than alpha radiation. This instrument, shown in Fig. II-2, is manufactured by the Eberline Instrument Company as their model PAC-4G meter with a probe.

Both of these instruments are calibrated at ORNL using ^{239}Pu alpha sources. While each instrument is individually calibrated, the calibration factors are typically 5-6 dpm/cpm.

ORNL-Photo 6707-76

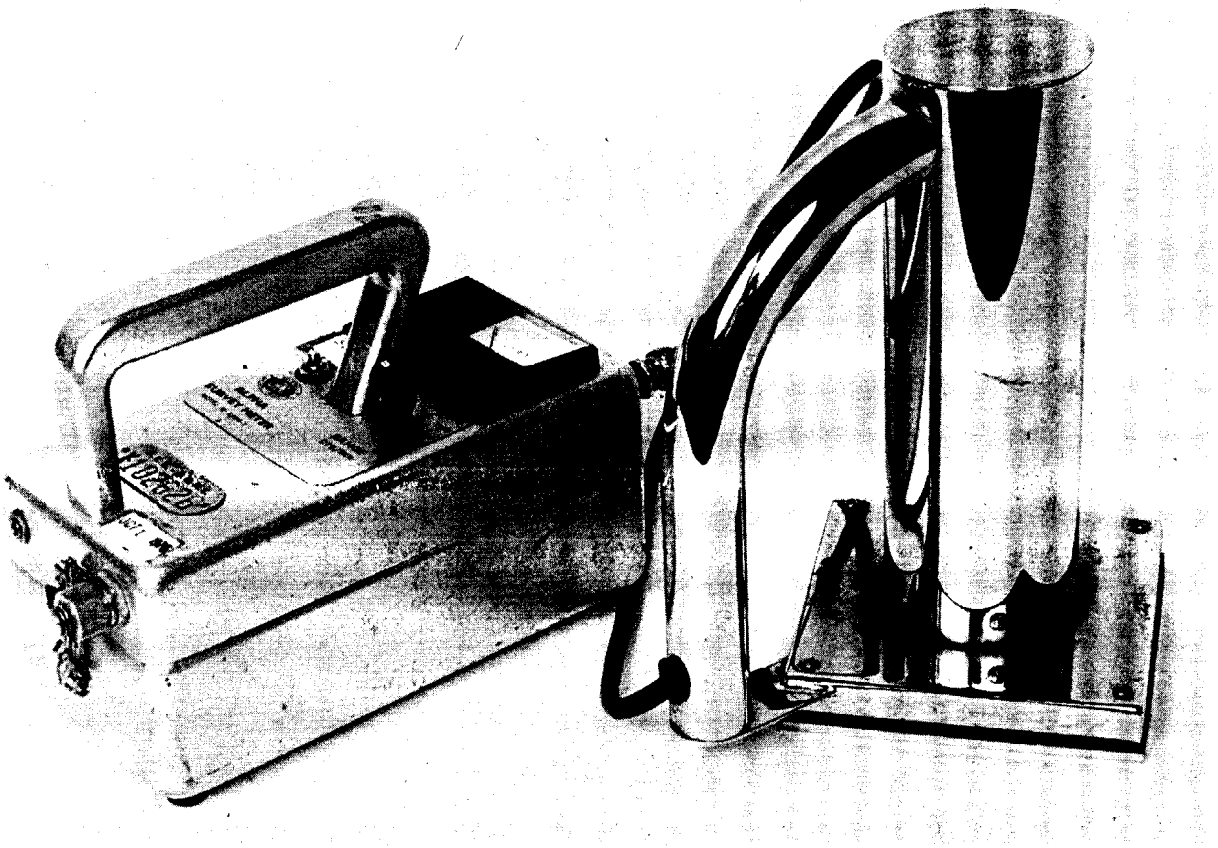

Fig. II-1. Alpha scintillation survey meter.

ORNL-Photo 6715-76

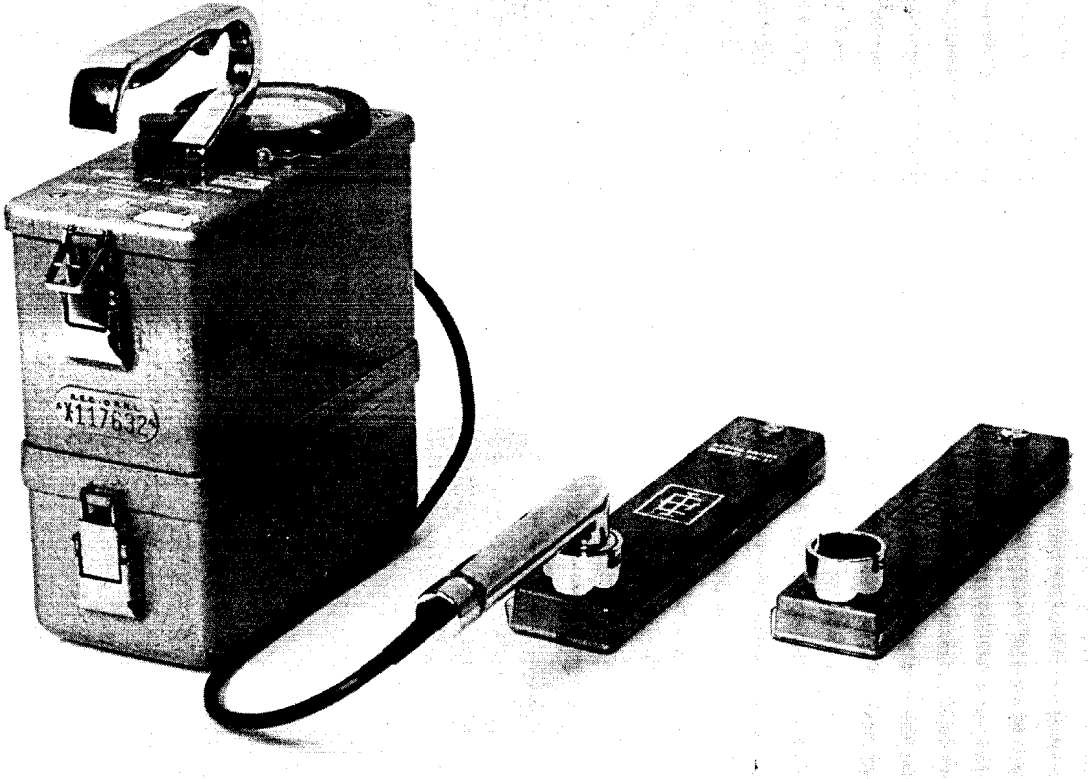

Fig. II-2. Gas-flow proportional alpha survey meter.

Beta 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. II-3.

The G-M survey meter was calibrated at ORNL for gamma radiation using an NBS standard radon source. The gamma calibration factor is typically of the order of 2,600 cpm mR/h.

In order to assess beta-gamma surface dose rates from uranium-contaminated surfaces using this instrument, a field calibration was performed. The G-M survey meter was compared with a Victoreen Model 440 ionization chamber (see Fig. II-4) and was found to produce 1,750 cpm per mrad/h with a 25% standard deviation for a wide variety of surfaces, including concrete, wood, pavement, bricks, and steel beams.

Gamma Scintillation Survey Meter

A portable survey meter using a NaI scintillation probe is used to measure low-level gamma radiation exposure. The scintillation probe is a 3.2 × 3.8-cm NaI crystal coupled to a photomultiplier tube. This probe is connected to a Victoreen Model Thyac III ratemeter (see Fig. II-5). This unit is capable of measuring radiation levels from a few μR/h to several hundred μR/h. This instrument is calibrated at ORNL with an NBS standard ^{226}Ra source. Typical calibration factors are of the order of 300 cpm per μR/h.

ORNL-Photo 6704-76

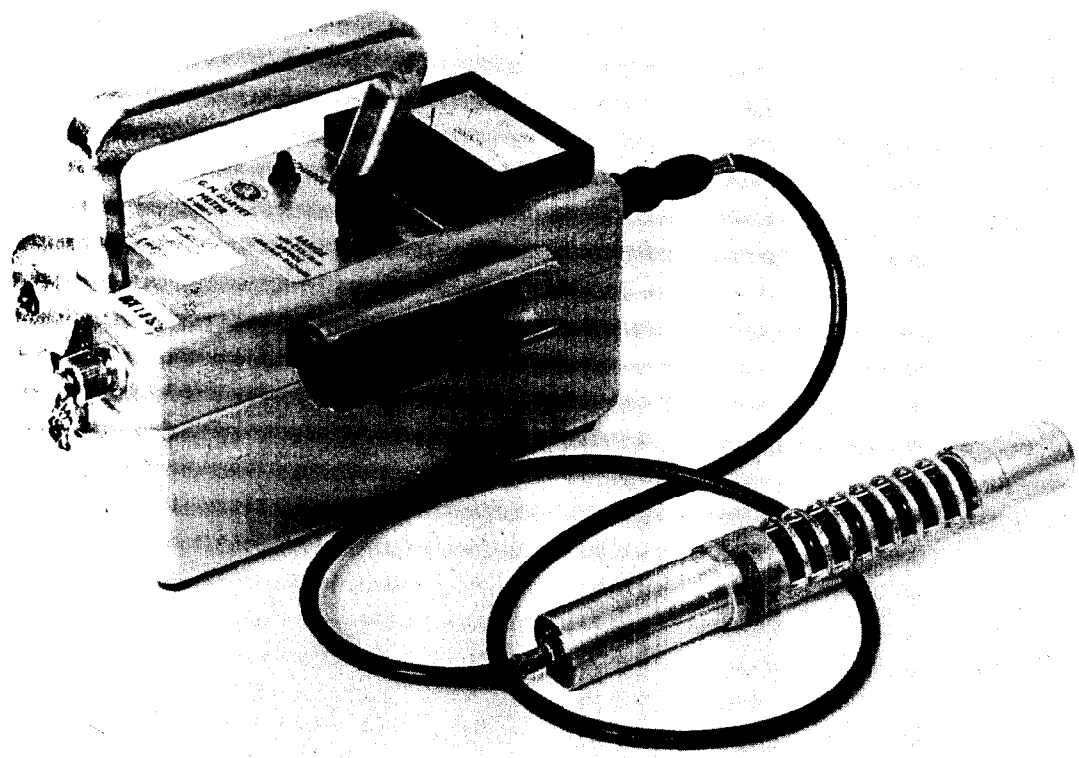

Fig. II-3. Geiger-Muller survey meter.

ORNL-Photo 6710-76

Fig. II-4. Victoreen model 440 ionization chamber.

ORNL-Photo 6705-76

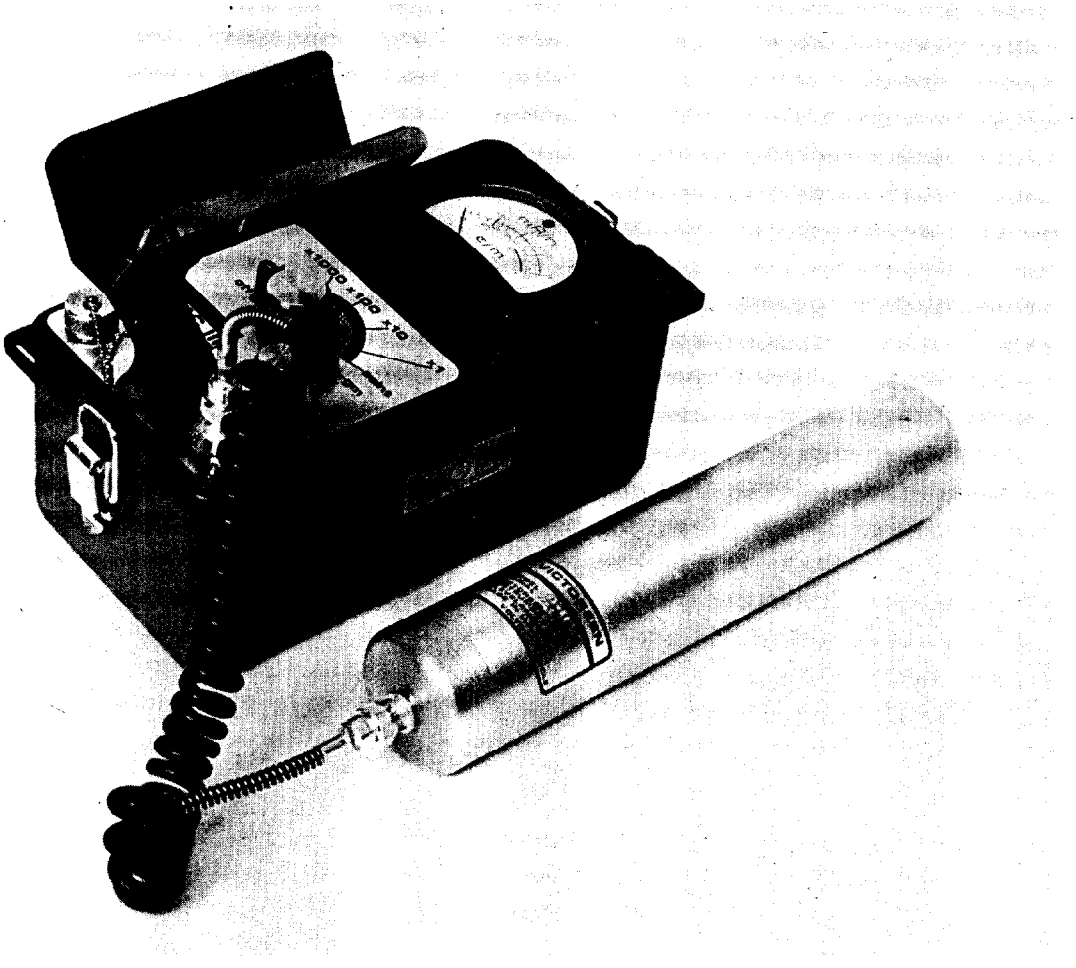

Fig. II-5. Victoreen model Thyac III rate-meter.

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. II-6). 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 $\times$ 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. II-6, was used in the field to measure beta activity on smear papers and was calibrated daily using a beta standard of known activity.

Fig. II-7 shows the mobile lab used during the survey.

ORNL-Photo 1070-78

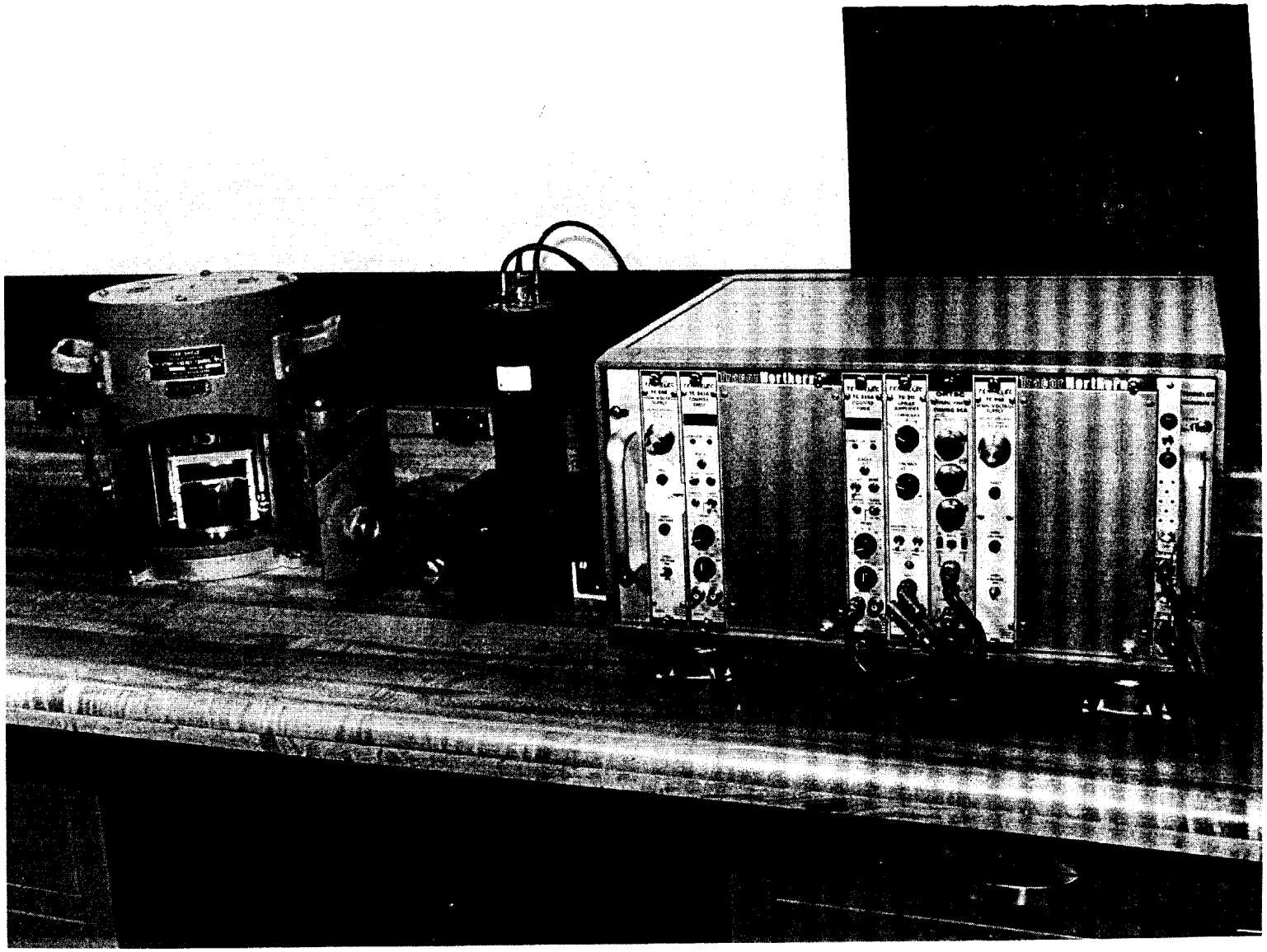

Fig. II-6. Alpha and beta smear counters.

ORNL-Photo 5838-76

Fig. II-7. Mobile lab used on survey.

APPENDIX III

DESCRIPTION OF Ge(Li) DETECTOR AND SOIL COUNTING PROCEDURES

DESCRIPTION OF Ge(Li) DETECTOR SYSTEM

A holder for twelve 30-cm^3 polyethylene bottles (standard containers for liquid scintillation samples) and a background shield have been designed for use with a 50-cm^3 Ge(Li) detector system (see Fig. III-1 and III-2). During counting of the samples, the holder is used to position 10 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.

Pulses are sorted by a 4096-channel analyzer (see Fig. III-3), stored on magnetic tape, and subsequently entered into a computer program which uses an iterative least-squares method to identify radionuclides corresponding to those gamma-ray lines found in the sample. The program, which is accessible through a remote terminal, relies on a library of radioisotopes which contains approximately 700 isotopes and 2,500 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, 1,120, 1,765, and 2,204 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

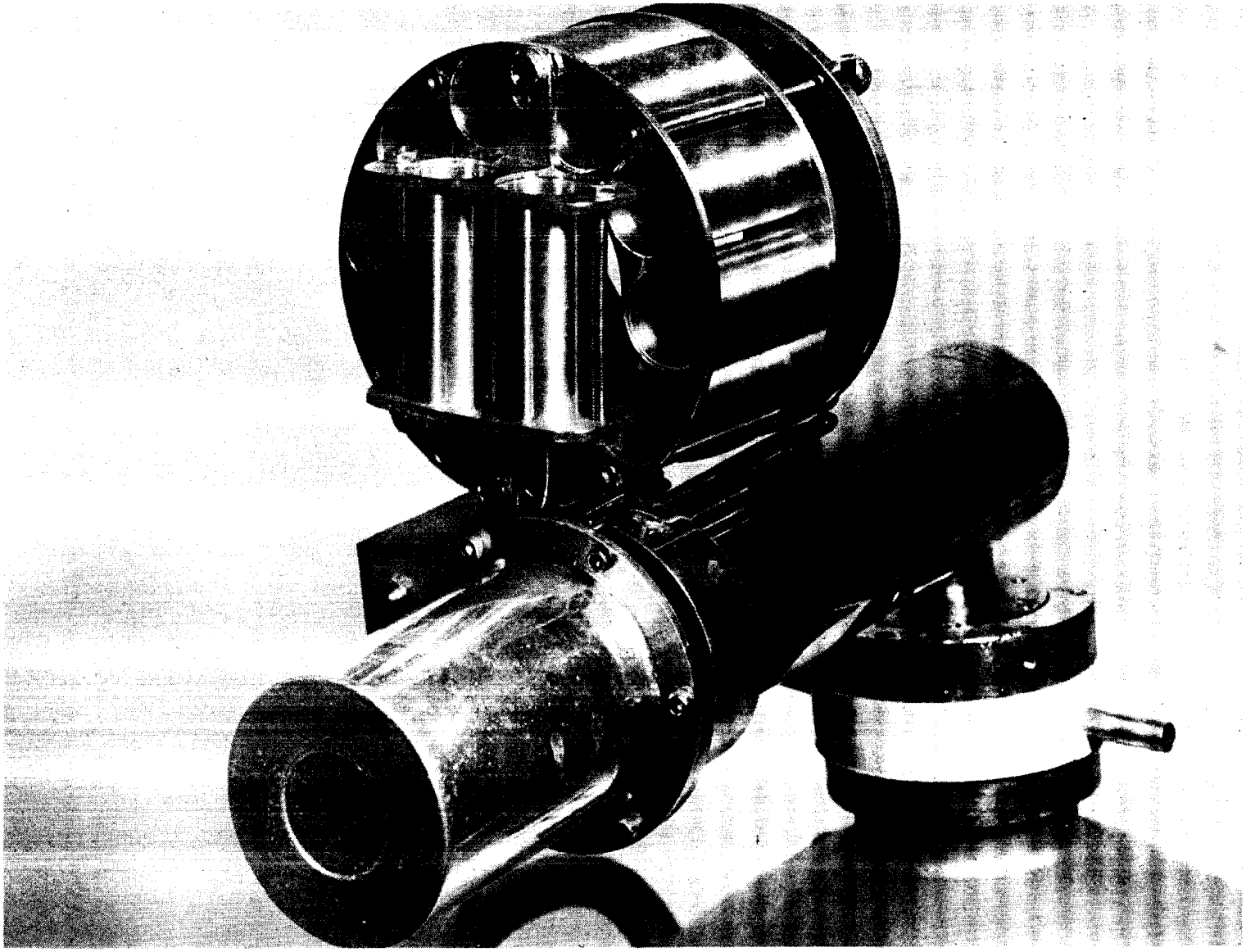

Fig. III-1. Holder for Ge(Li) detector system.

ORNL-Photo 2171-75

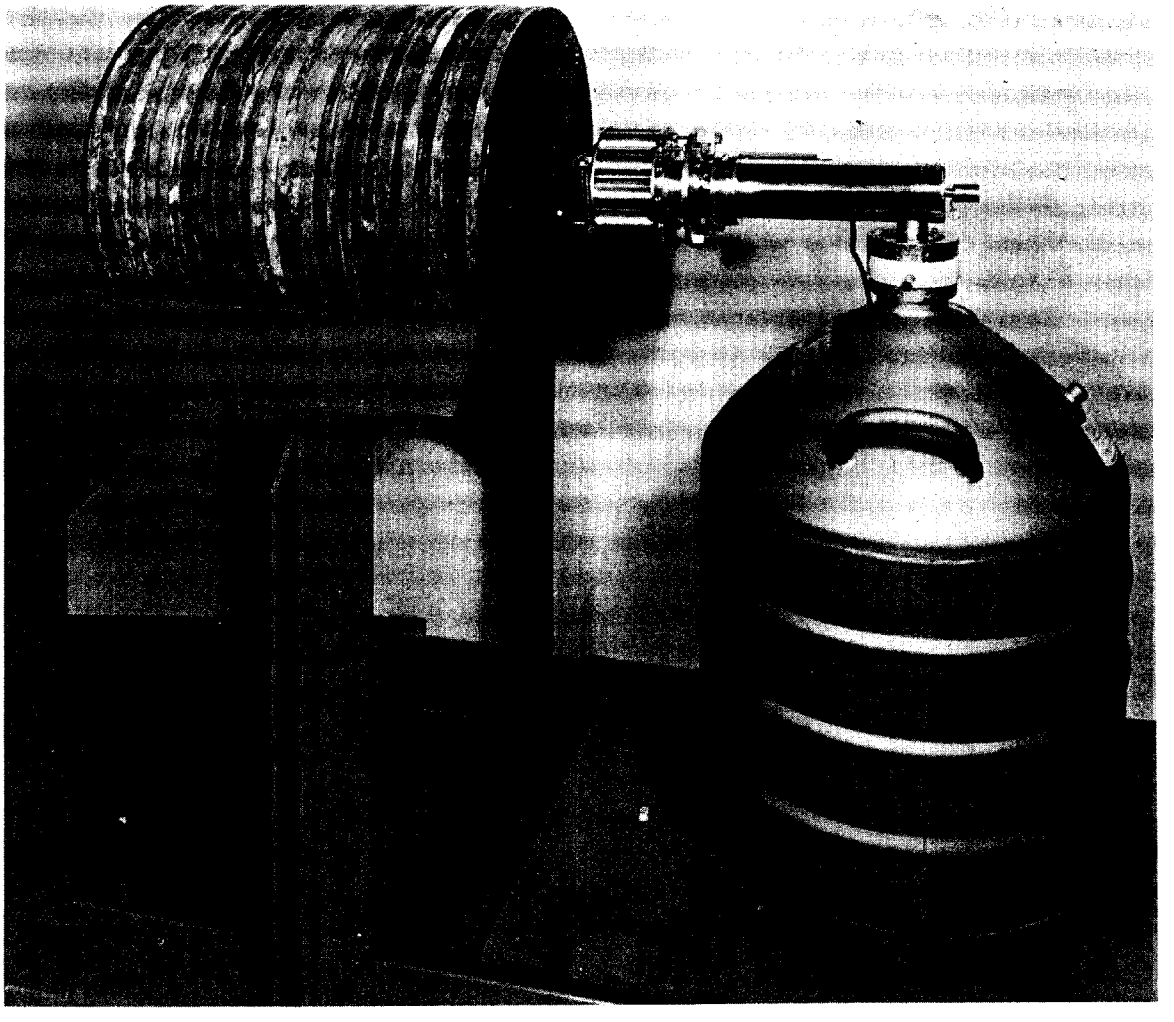

Fig. III-2. Ge(Li) detector system.

ORNL-Photo 6711-76

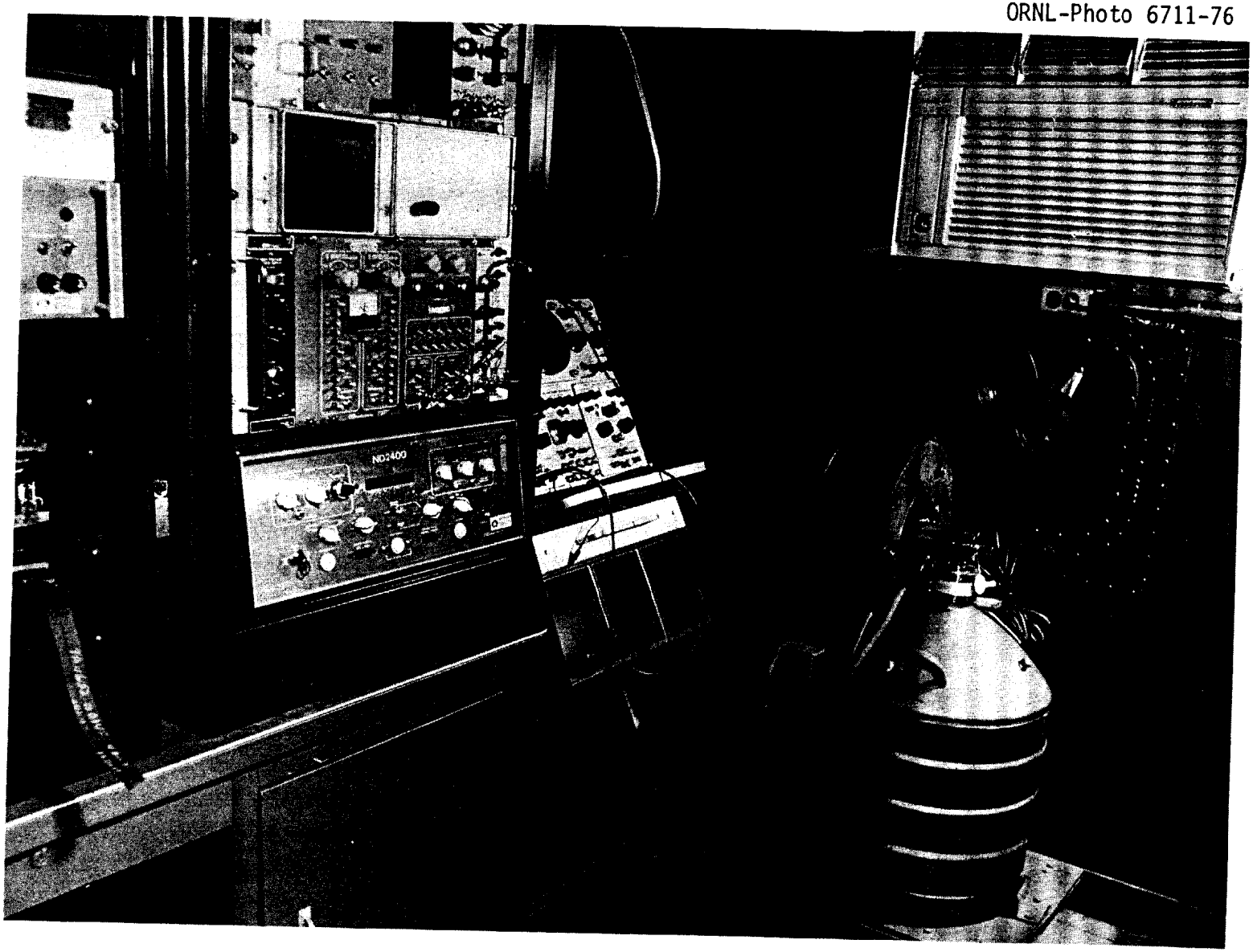

Fig. III-3. 4096 - channel analyzer.

APPENDIX IV

PROCEDURE FOR ESTIMATING THE ^{238}U CONTAMINATION FROM SCINTILLATION PROBE LOGGINGS

PROCEDURE FOR ESTIMATING THE ^{238}U CONTAMINATION FROM SCINTILLATION PROBE LOGGINGS

Two of the immediate daughters of ^{238}U are gamma-ray emitters: ^{234}Th with 63 keV at 5% yield and a 93 keV complex at 5% yield, and ^{234m}Pa with 1,001 keV at 0.6% yield. The half-lives of these daughters are short enough that equilibrium with the freshly separated ^{238}U occurs within a few months. Thus, gamma-ray counting could provide an estimate of ^{238}U activity. A large fraction of the low-energy (63 keV and 93 keV) gamma rays can be absorbed by a medium such as sand or soil and not be detected by the NaI scintillator. Furthermore, gamma rays associated with the daughters of ^{226}Ra have a higher probability of interaction with the scintillator than does the 1,001 keV gamma ray. Because of this higher efficiency of detection and the higher yield of ^{226}Ra daughter gamma rays, the presence of background levels of ^{226}Ra can cause appreciable interferences in estimating ^{238}U activity on the basis of gamma-ray counting.

Soil and crud samples were obtained from holes drilled in the service pits at various locations. The results of ^{238}U analysis for 10 of these samples could be compared with the shielded and unshielded gamma-ray loggings of the holes. "Best-fitting" lines were obtained for these relationships as follows:

$$^{238}U\ (pCi/g) = 3.07 \times 10^{-3} \times (\text{unshielded cpm}) - 2.23,$$

$$^{238}U\ (pCi/g) = 20.33 \times 10^{-3} \times (\text{shielded cpm}) - 7.55.$$

Use of either of these relationships can yield estimates which may be in error by 200% because of the low gamma-ray yield and ^{226}Ra interference problems outlined above.

Estimates of ^{238}U concentration given in Table 12 were made by using these two relationships and the count-rate data from the hole loggings. The average of the two estimates obtained (unshielded and shielded) is given in Table 13 for each depth logged.

APPENDIX V

EVALUATION OF RADIATION EXPOSURES AT THE FORMER BRIDGEPORT BRASS COMPANY SPECIAL METALS EXTRUSION PLANT, ADRIAN, MICHIGAN

The U.S. Department of Energy (DOE) has determined that the former Bridgeport Brass Company Special Metals Extrusion Plant at Adrian, Michigan, is presently contaminated with low-level radioactive residues resulting from previous uses of this property. This plant was operated under contract with the Atomic Energy Commission (AEC) from 1953 to 1962 to extrude uranium metal which could be used to produce fuel for AEC reactors at Hanford, Washington, and Savannah River, South Carolina. Only about 40,000 square feet of one building of the 17.4-acre structure was involved in the AEC contract work. When the contract was terminated in 1962, one of the plant's two extrusion presses was sent to Reactive Metals, Inc., Ashtabula, Ohio, and put into operation there. Other equipment was scrapped and disposed of at unknown locations.

In 1974, the Chevrolet Manufacturing Division of General Motors purchased the site, cleared the buildings, and conducted a radiological survey. A comprehensive radiological survey performed by DOE indicated that some low-level uranium contamination from AEC-contracted work remains at this site in spite of extensive decontamination work done by the current property owner. This contamination from previously contracted operations is producing radiation exposures which are, for all practical purposes, indistinguishable from natural background. These slight exposures to employees working at this site result from beta and gamma radiation emitted by contamination in the ground or on normally inaccessible surfaces. A summary of radiation exposures at the former Bridgeport Brass site is provided in Table V-1 along with appropriate guidelines and background values.

The naturally occurring radionuclides present at the former Bridgeport Brass site are also present in minute quantities throughout our environment. Concentrations of these radionuclides in normal soils, 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.

Table V-1. Summary of exposure data at the former Bridgeport Brass, Adrian, Michigan

Exposure source	Background levels	Guideline value for general public	Guideline value for radiation workers	Average levels at Bridgeport Brass site
Gamma radiation from daughters of uranium contamination	9 microRoentgens per hour in Michigan	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 were 12 to 13 microRoentgens per hour
Beta-gamma radiation from daughters of uranium contamination	Less than 0.02 millirad per hour	Maximum of 0.2 millirad per hour at 1 centimeter above surface of 1 square meter	15 millirads per hour for 40 hours per week and 50 weeks per year. This is equivalent to 30 rads per year	Range from background to 5 millirads per hour. Guideline values never exceeded at any accessible point indoors or outdoors

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.

The rad is the unit for measuring radiation dose to tissue. One millirad is equal to one-thousandth of a rad.

The use of radioactive materials for scientific, industrial, or medical purposes may cause radiation exposure 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, there would only be one-fourth curie of uranium-238. 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 V-2 until stable lead is formed. The decay product listed in Table V-2 is the radiation produced as the parent decays.

Direct Gamma-Ray Exposures

As shown in Table V-2, some of the radioactive daughters in the uranium decay series emit gamma-rays. (Gamma-rays are a highly penetrating radiation like X-rays.) The maximum gamma radiation level indoors was recorded at an isolated spot at the south end of the building and was equal to 31 microRoentgens per hour.† The average indoor reading was 13 microRoentgens per hour. Outdoor readings ranged to 15 microRoentgens per hour, with an average value of 12 microRoentgens per hour. Exposure to the maximum indoor gamma radiation level, 31 microRoentgens per hour

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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.

†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.

Table V-2. Uranium 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	2,600 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	140 days	alpha	lead-206
lead-206	stable	none	none

[a]Short-lived radon daughters.

for 2,000 hours per year (a typical work year), would lead to an exposure of 62,000 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.

On the basis of information provided by the U.S. Environmental Protection Agency (EPA), it is estimated that the average background exposure in Michigan is about 9 microRoentgens per hour.* Since local variation of up to 50% in background radiation is common, it may be concluded that both the indoor average and the outdoor average gamma radiation levels cannot be distinguished from background at this site.

The National Council on Radiation Protection and Measurements (NCRP) 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. This guideline would not be exceeded at any accessible location on this site.

Direct Beta-Gamma Exposures

Nuclear Regulatory Commission (NRC) guidelines state that the combined dose from weakly penetrating beta particles and from gamma rays, measured at a distance of 1 centimeter above any surface, should not exceed 0.2 millirad† per hour when averaged over an area of 1 square meter. The combined dose rate should not exceed 1.0 millirad per hour in small areas of 100 square centimeters. These guidelines are not exceeded in any routinely accessible areas of the building nor at a point outdoors. However, the guidelines are exceeded at some locations inside drains, manholes, and sumps. The maximum value obtained was 5 millirad per hour inside manhole 16. Handling material inside this manhole for one hour would result in a skin dose of 5 millirad. For comparison, the skin dose which would be expected from a normal year's

*D. T. Oakley, *Natural Radiation Exposure in the United States*, U.S. Environmental Protection Agency Publication ORP/S1O, 72-1, June 1972.

†The rad is the unit for measuring radiation dose to tissue. One millirad is equal to one-thousandth of a rad.

watching of color television by an adult is 1.6 millirad; 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 1 foot away from these surfaces. Although potential exists for exposures in excess of the guidelines, beta and gamma exposures are believed to be inconsequential to employees at this site due principally to the routine inaccessibility of the surfaces exceeding the guidelines.

Other Considerations of Exposure

Several sumps, drains, and manholes are contaminated with uranium-238 residues. For example, a sample of material from the bottom of manhole M-16 had a uranium-238 concentration of 15,000 picoCuries* per gram (normal soil has about 1 picoCurie per gram). Furthermore, the sand in the service pits below the floor had up to 200 picoCuries of uranium-238 per gram. These materials are capable of causing human skin exposures by improperly handling the material. Additional, more serious inhalation exposures could result from improper removal of this material in a manner which could cause it to become airborne. Ingestion exposures could also conceivably result if this material were disposed of in a manner in which any member of the public would have access to it.

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:

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*A picoCurie is one millionth-millionth of a Curie, previously defined.

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.

Any 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, an individual who smokes cigarettes has subjected himself to a risk of lung cancer which is about 10 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 cancer deaths in the relatively small working population at the former Bridgeport Brass site.

The normal annual death rate* from all types of cancer among all population groups in Lenawee County (as of 1970) was 154 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 Michigan was 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, any action taken to reduce either the rate or the duration of radiation exposures would also reduce the risk attendant to that exposure.

There are no data at present which give evidence of a relationship between low-level exposure of the skin and the development of skin cancers. This does not mean that skin cancer cannot be produced by low-level exposures. This does mean that the risk associated with guideline level exposures of the skin is so small that it cannot be quantified.

Remedial Measures

The present radiation exposures at the former Bridgeport Brass facility at Adrian, Michigan, may be considered trivial. However, the presence of uranium-238 contamination in inaccessible places warrants consideration of remedial action to prevent future radiation exposures. Thoroughly cleaning the drains, manholes, sumps, and service pits under controlled conditions and proper disposal of the contaminated residue would, of course, provide maximum protection. Stabilization or fixation of the material in place and restricting future activities of the site could also be considered. The Department of Energy is now actively evaluating alternatives under a priority program designed to assure protection of the working population.

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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.

SUMMARY

In summary, the former Bridgeport Brass Company Special Metals Extrusion Plant at Adrian, Michigan, is contaminated with residues containing naturally occurring radionuclides. Radiation exposures associated with this contamination under conditions of current use cannot be distinguished from background. The present property owner has adequately decontaminated the readily accessible portions of this site. However, the contamination in inaccessible portions of the site has the potential to produce radiation exposures which could approach and, in some cases, exceed scientifically based guidelines. These potentially more serious exposures could result from changes in accessibility to the contaminated material. 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 Bridgeport Brass facility. Currently, work is underway to implement the elements of this plan.

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