CONFIRMATORY SURVEY OF THE U.S. ARMY MATERIALS TECHNOLOGY LABORATORY RESEARCH REACTOR FACILITY WATERTOWN, MASSACHUSETTS

T. J. VITKUS

Prepared for the
Division of Industrial and Medical Nuclear Safety
U.S. Nuclear Regulatory Commission
Region I Office

ORISE

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

Environmental Survey and Site Assessment Program
Energy/Environment Systems Division

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NOTICES

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Prepared by

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FINAL REPORT

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ABBREVIATIONS AND ACRONYMS

AMTL	U.S. Army Materials Technology Laboratory
ASME	American Society of Mechanical Engineers
cm^2	square centimeter
cpm	counts per minute
DOE	U.S. Department of Energy
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
EML	Environmental Measurements Laboratory
EPA	Environmental Protection Agency
ESSAP	Environmental Survey and Site Assessment Program
ft	foot
GM	Geiger-Müeller
kg	kilogram
km	kilometer
m	meter
m^2	square meter
MeV	megaelectron volts
MK	Morrison Knudson Corporation
mm	millimeter
MW	megawatt
NaI	Sodium Iodide
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
μR/h	microroentgens per hour
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
PIC	pressurized ionization chamber
QA	Quality Assurance
RCA	Radiation Control Area
SEG	Scientific Ecology Group, Inc.
ZnS	Zinc Sulfide

CONFIRMATORY SURVEY
OF THE
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RESEARCH REACTOR FACILITY
WATERTOWN, MASSACHUSETTS

INTRODUCTION

The U.S. Army Materials Technology Laboratory (AMTL) Research Reactor Facility was operated for the Department of Defense from June 1960 until March 1970. Reactor operations were conducted under Atomic Energy Commission (AEC), predecessor organization to the U.S. Nuclear Regulatory Commission (NRC), license Number R-65 (Docket File No. 50-47). The reactor was originally a one MW pool-type materials testing reactor, but the license was amended twice to provide for power level increases to two MW and five MW respectively. Experiments conducted included solid-state physics research programs by the Army Materials and Mechanics Research Center and local universities performed diffraction measurements and irradiations. Several leaks of coolant water from the concrete pool wall were noted, as was a release from a concrete holding tank (Cistern 242) during the reactor operating history.

A deactivation report was submitted to the AEC, predecessor organization to the NRC, in December 1970, after reactor operations ended. The fuel, some reactor components, and other contaminated and non-contaminated equipment and materials were then removed. A characterization survey of the facility was conducted in 1990; this was followed by submittal of a decommissioning plan to the NRC. Contaminants identified included cobalt-60, europium-152, europium-154, and cesium-137. Decommissioning began in June 1992, and completed in January 1993 with the Morrison Knudson Corporation (MK) serving as the prime contractor. Decommissioning activities included vacuuming and excavation of interior and exterior pipe and drain systems, Cistern 242, and dismantlement of equipment and structural components; jackhammering and scabbling surfaces; and removal of the secondary cooling system. The Scientific Ecology Group, Inc. (SEG), under contract to MK, provided health physics services during the decommissioning and then performed the final radiological survey, the results of which were published in a March 1993 final survey report.[1]

This decommissioning project was undertaken in order to terminate the NRC license and release the reactor facilities to unrestricted use as the AMTL base is scheduled for closure in 1995. At the request of the NRC's Region I Office, the Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) performed an independent survey of the AMTL Research Reactor Facility.

SITE DESCRIPTION

The AMTL was constructed in the late 1950s on the site of the old Watertown Arsenal in Watertown, Massachusetts which is approximately 14 km (9 mi) west of Boston's Logan Airport (Figure 1). The Research Reactor Facility is situated in the south-central portion of the AMTL and is enclosed within a 55 m (180 ft) x 98 m (320 ft) fenced area known as the reactor yard (Figure 2). The yard is bordered by Wooley Ave. to the north, Craig St. to the east, N. Beacon St. to the south, and Building 97 and other AMTL property to the west (Figure 3). The Reactor Facility was comprised of Building 100, which housed the reactor, the radiation control area (RCA) within the adjacent Building 97, Cistern 242, and a secondary cooling system. Prior to the decommissioning, Building 100 also housed the reactor control room, fuel storage racks, primary cooling system, a gamma ray experimental facility, and the electrical and mechanical service centers.

The reactor monolith occupied the center of circular Building 100, rested on the basement floor, and extended upwards for three levels. Building construction is of concrete with a steel plate shell. Decommissioning activities have essentially gutted the interior of the building leaving only about 600 m^2 of the basement and first floor area, airlocks, equipment hatch, the reactor dome walls, and basement walls (Figures 4 through 6). In addition, pipe and drain removal left a number of trenches in the basement floor. The basement floor also contained the 5 m deep gamma well and a sump. At the time of the confirmatory survey, groundwater had percolated into the trenches, gamma well, and sump.

The reactor yard is principally a soil surface and contained the Cistern, pipe chases and secondary cooling system excavations, which have been backfilled, and resultant soil piles. The concrete pad, that at one time supported exhaust stacks, is also in the reactor yard.

OBJECTIVES

The objectives of the confirmatory process were to provide independent document reviews and radiological data, for use by the NRC in evaluating the adequacy and accuracy of the licensee's final status report, relative to established guidelines.

DOCUMENT REVIEW

ESSAP reviewed the licensee's final radiological survey report.[1] Procedures and methods utilized by the licensee were reviewed for adequacy and appropriateness. The data was reviewed for accuracy, completeness, and compliance with guidelines.

PROCEDURES

An ESSAP survey team visited the AMTL during the period of April 26 through 29, 1993 and performed visual inspections and independent measurement and sampling of the Research Reactor Facility. The survey was conducted in accordance with a plan dated April 20, 1993 submitted to and approved by the NRC Region I.[2] This report summarizes the procedures and results of the survey.

SURVEY PROCEDURES: INTERIOR

The following procedures were used for the confirmatory survey of the Building 100 interior.

Reference Grid

The 3 ft x 3 ft reference grid systems established by the licensee on the floor and lower walls were utilized for survey reference.

Surface Scans

Alpha, beta, and gamma surface scans were performed on the operating deck, basement, air locks and RCA floors and lower walls using large area gas proportional and NaI detectors. The basement trenches, sump, and gamma well were scanned using GM and NaI detectors. All detectors were coupled to countrate meters with audible indicators. Locations of elevated direct radiation identified by scans were marked for further investigation.

Surface Activity Measurements

Total and removable alpha and beta activity measurements were performed in 110 randomly selected floor and lower wall grid blocks. One "set" of five direct measurements was obtained from each grid block selected for survey. Additional single-point direct measurements were made within the gamma well, air locks, equipment hatch, and pipe penetrations. Direct measurements were made using GM and ZnS detectors, coupled to ratemeter-scalers. Removable activity levels were determined by taking a smear for each set of five direct measurements corresponding to the location of the highest total activity, and from each single-point measurement location. Figures 7 through 11 indicate measurement locations.

Exposure Rates

The interior background and Building 100 exposure rate measurement data, developed by SEG, was reviewed for adequacy, accuracy, and guideline compliance. In addition, ESSAP monitored the gamma activity levels during surface scans in order to identify any areas of elevated direct gamma radiation in excess of background.

Soil Sampling

Trenching operations in the basement level did not penetrate the concrete foundation to the underlying soils. Therefore, soils were not accessible for sampling.

SURVEY PROCEDURES: EXTERIOR

The following procedures were used during the radiological survey of the reactor yard.

Reference System

ESSAP referenced measurement and sampling locations to prominent site features.

Surface Scans

Gamma surface scans were performed over 100 percent of the reactor yard surfaces using NaI detectors coupled to ratemeters with audible indicators.

Surface Activity Measurements

One set of five direct measurements for total alpha and total beta activity levels was performed on the reactor yard stack pad (Figure 12). Measurements were performed using ZnS and GM detectors coupled to ratemeter-scales. A smear sample was collected from the measurement location corresponding to the location of highest total activity.

Soil Sampling

Background soil samples were collected from 6 locations within 0.5 to 5 km of the site (Figure 13). Surface soil samples were collected from 4 locations within the reactor yard (Figure 14). In addition, ESSAP selected 10 of the licensee's remedial action samples, which

SEG had collected from the excavations and overburden piles, and the NRC provided 5 soil samples from the Cistern 242 excavation for confirmatory analysis.

Exposure Rate Measurements

Background exposure rate measurements were made at each of the background soil sampling locations (Figure 13). Site exposure rate measurements were made at 4 locations within the reactor yard (Figure 12). All exposure rates were measured at 1 m above the surface using a pressurized ionization chamber (PIC).

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ORISE's ESSAP laboratory in Oak Ridge, TN for analysis and interpretation. Smears were analyzed for gross alpha and gross beta activity using a low background gas proportional counter. Direct measurement and smear data were converted to units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2). Soil samples were analyzed by gamma spectrometry. The spectra were reviewed for Co-60, Cs-137, Eu-152, Eu-154, uranium, and any other identifiable photopeaks. The analytical results for soil samples were reported in pCi/g. Exposure rates were reported in μR/h. The data generated was compared with the licensee's documentation and NRC guidelines established for release to unrestricted use.

FINDINGS AND RESULTS

DOCUMENT REVIEW

Comments identified in the AMTL final survey report were provided to the NRC in an April 16, 1993 correspondence.[3] Resolution of the issues identified were adequately addressed by the licensee in the revised final survey report provided by SEG.[1] The document, therefore, provides an adequate description of the radiological condition of the building relative to the NRC's requirements for release to unrestricted use.

INTERIOR

Surface Scans

Alpha, beta, and gamma surface scans of the floors and lower walls of Building 100, the RCA of Building 97, the airlocks, equipment hatch, gamma well, trenches, sump, and pipe penetrations did not identify any locations of elevated direct radiation.

Surface Activity Levels

Results of total and removable activity levels are summarized in Table 1. The average activity levels in 1 m^2 grid blocks were less than 83 dpm/100 cm^2 for alpha and less than 1500 dpm/100 cm^2 for beta. Total activity levels for single measurements were less than 83 dpm/100 cm^2 for alpha and ranged from less than 1400 to 1500 dpm/100 cm^2 for beta. Removable activity levels were less than 12 dpm/100 cm^2 and less than 17 dpm/100 cm^2 for gross alpha and gross beta, respectively.

Exposure Rates

SEG determined the average interior background exposure rate range to be 13.2 to 20.3 μR/h. SEG's average exposure rate measurements within Building 100 were 4.9 to 15.2 μR/h. ESSAP did not identify any gamma activity which exceed background levels during confirmatory surface scans.

EXTERIOR

Surface Scans

Gamma surface scans of the reactor yard did not identify any locations of elevated direct radiation.

Exposure Rates

Background exposure rate measurements are summarized in Table 2 and ranged from 10 to 11 μR/h. Exposure rates in the Reactor Yard ranged from 11 to 12 μR/h and are summarized in Table 3.

Radionuclide Concentrations in Soils

Background radionuclide concentrations in soils are summarized in Table 2. Concentration ranges were as follows: Co-60, less than 0.2 pCi/g; Cs-137, 0.1 to 0.7 pCi/g; Eu-152, less than 0.3 pCi/g; Eu-154, less than 0.5 pCi/g; U-235, 0.1 pCi/g; U-238, 0.4 to 1.1 pCi/g. Radionuclide concentrations in reactor yard soils are summarized in Table 4 and were: Co-60, less than 0.1 to 0.1 pCi/g; Cs-137, less than 0.1 to 0.3 pCi/g; Eu-152, less than 0.3 pCi/g; Eu-154, less than 0.5 pCi/g; U-235, 0.1 to 0.2 pCi/g; U-238, 0.5 to 2.0 pCi/g.

Surface Activity Levels

Surface activity levels on the exhaust stack pad were less than 73 dpm/100 cm^2 for alpha and less than 1500 dpm/100 cm^2 for beta (Table 1). Removable activity was less than 12 and less than 17 for gross alpha and gross beta, respectively.

COMPARISON OF RESULTS WITH GUIDELINES

The NRC's Regulatory Guide 1.86 establishes the guidelines for acceptable surface contamination levels used by the NRC to determine whether a licensed facility may be released to unrestricted use. These guidelines are provided in Appendix C. The licensee identified beta-gamma emitters, specifically Co-60, Cs-137, Eu-152, and Eu-154, as contaminants. The applicable guidelines that were used for data comparison for these contaminants are as follows:

Total Activity

5,000 β-γ dpm/100 cm^2, averaged over 1 m^2

15,000 β-γ dpm/100 cm^2, maximum in a 100 cm^2 area

Removable Activity

1,000 β-γ dpm/100 cm^2

The licensee chose to compare alpha direct measurement data to the most conservative guidelines for alpha emitters which are:

Total Activity

100 dpm/100 cm^2, averaged over 1 m^2

300 dpm/100 cm^2, maximum in a 100 cm^2 area

Removable Activity

20 dpm/100 cm^2

All total and removable surface activity levels were well within these guideline values.

The exposure rate guideline applied to the site was 5 μR/h above background.[4] The exposure rates within Building 100 and the reactor yard were within this guideline.

The site-specific comparison value for residual radionuclide concentration in soil was the mean background level plus three standard deviations.[1] Residual concentrations of radionuclides were within the range of background concentration levels.

SUMMARY

The Environmental Survey and Site Assessment Program of the Oak Ridge Institute for Science and Education performed a confirmatory survey of the former Research Reactor Facility located at the Army Material Testing Laboratory in Watertown, Massachusetts during the period April 26 through 29, 1993. Survey activities included document reviews, surface scans, surface activity measurements, exposure rate measurements, and soil sampling.

The licensee's contractor-prepared final survey report adequately describes the radiological condition of the facility relative to residual contamination levels and exposure rates. The findings of the ESSAP survey support those of the termination survey contractor, SEG, and in ESSAP's opinion, indicate that the NRC guidelines for release to unrestricted use have been met for Building 100 and the reactor yard.

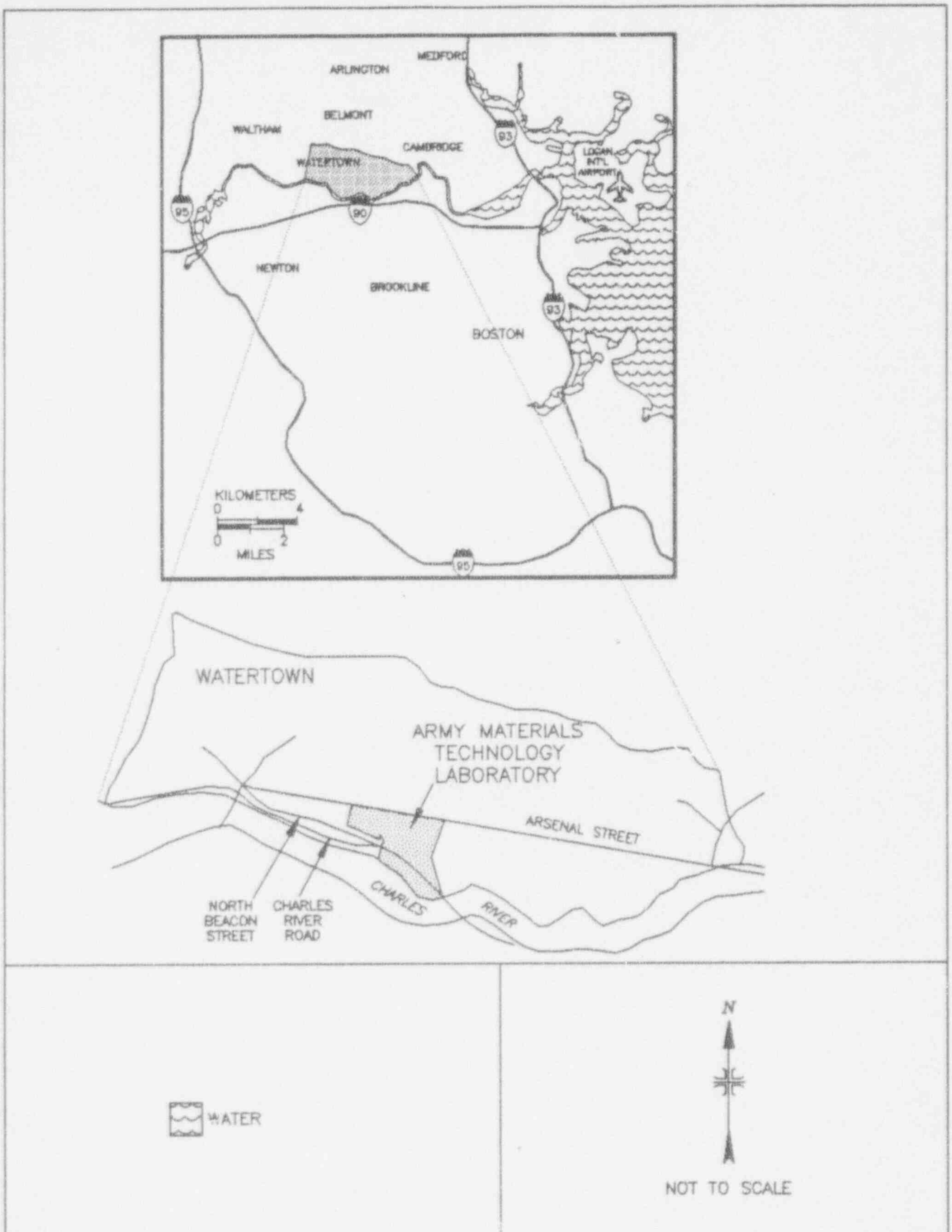

FIGURE 1: Location of the Army Materials Technology Laboratory, Watertown, Massachusetts

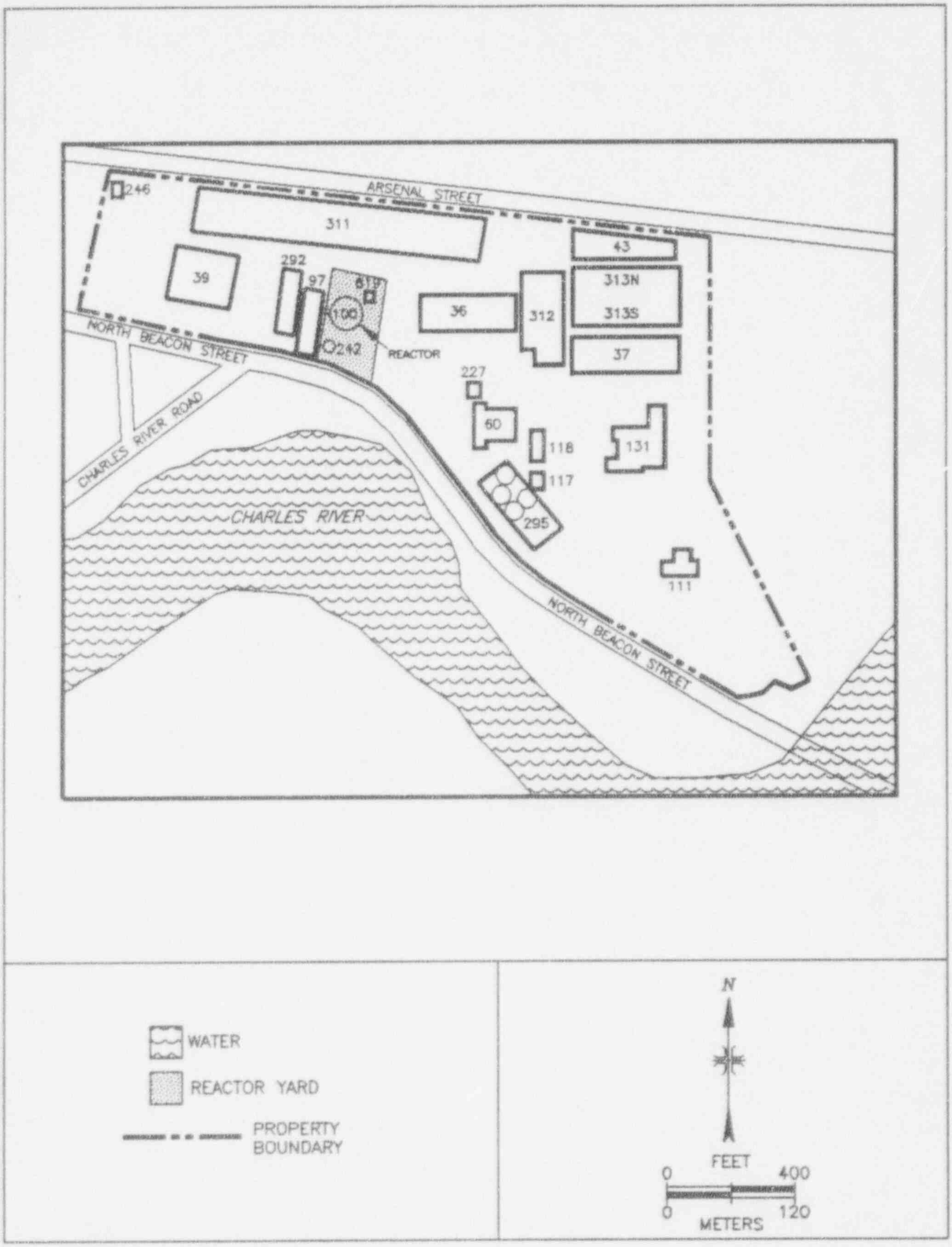

FIGURE 2: Location of the Army Materials Technology Laboratory Facilities

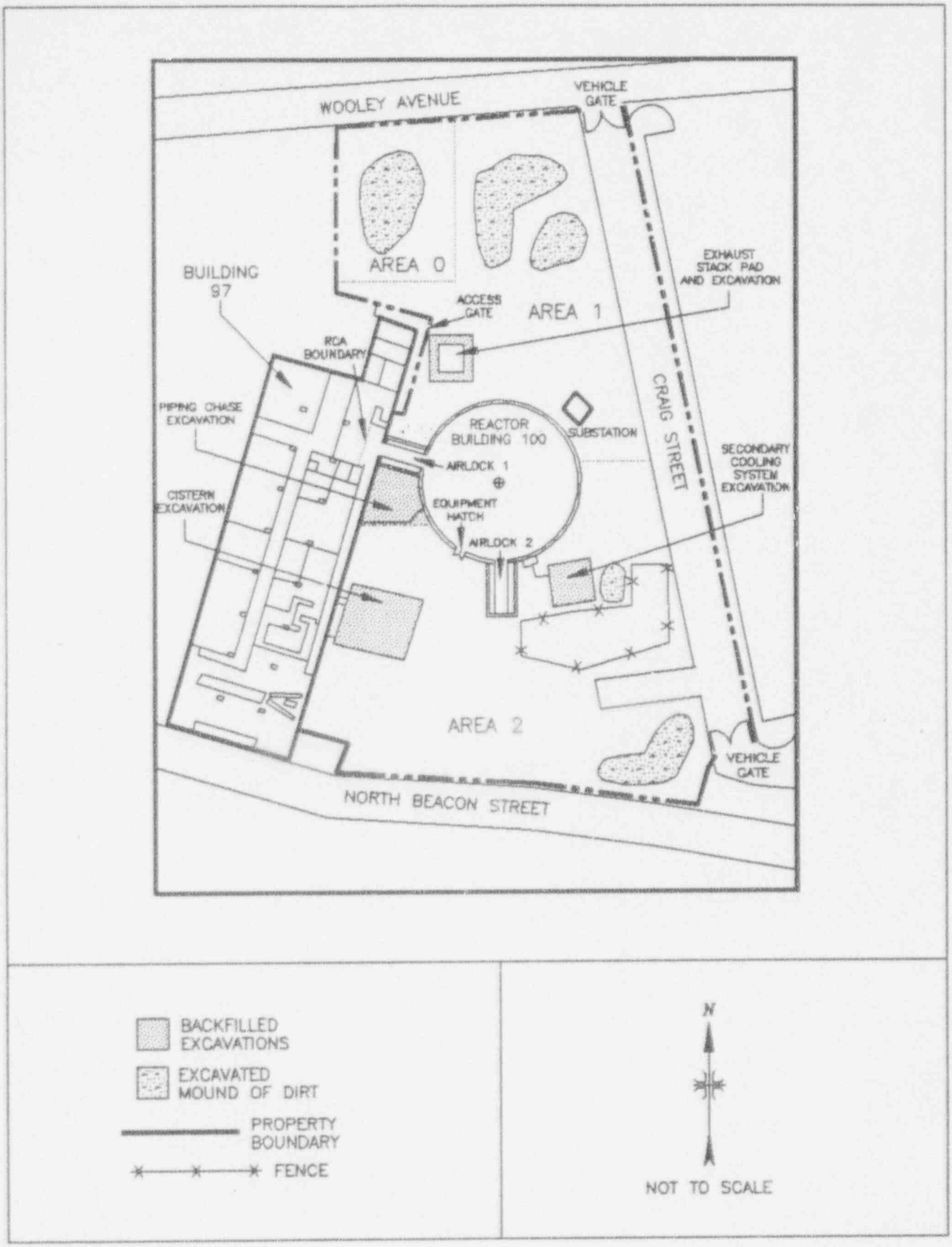

FIGURE 3: Plot Plan of Army Materials Technology Laboratory, Reactor Yard

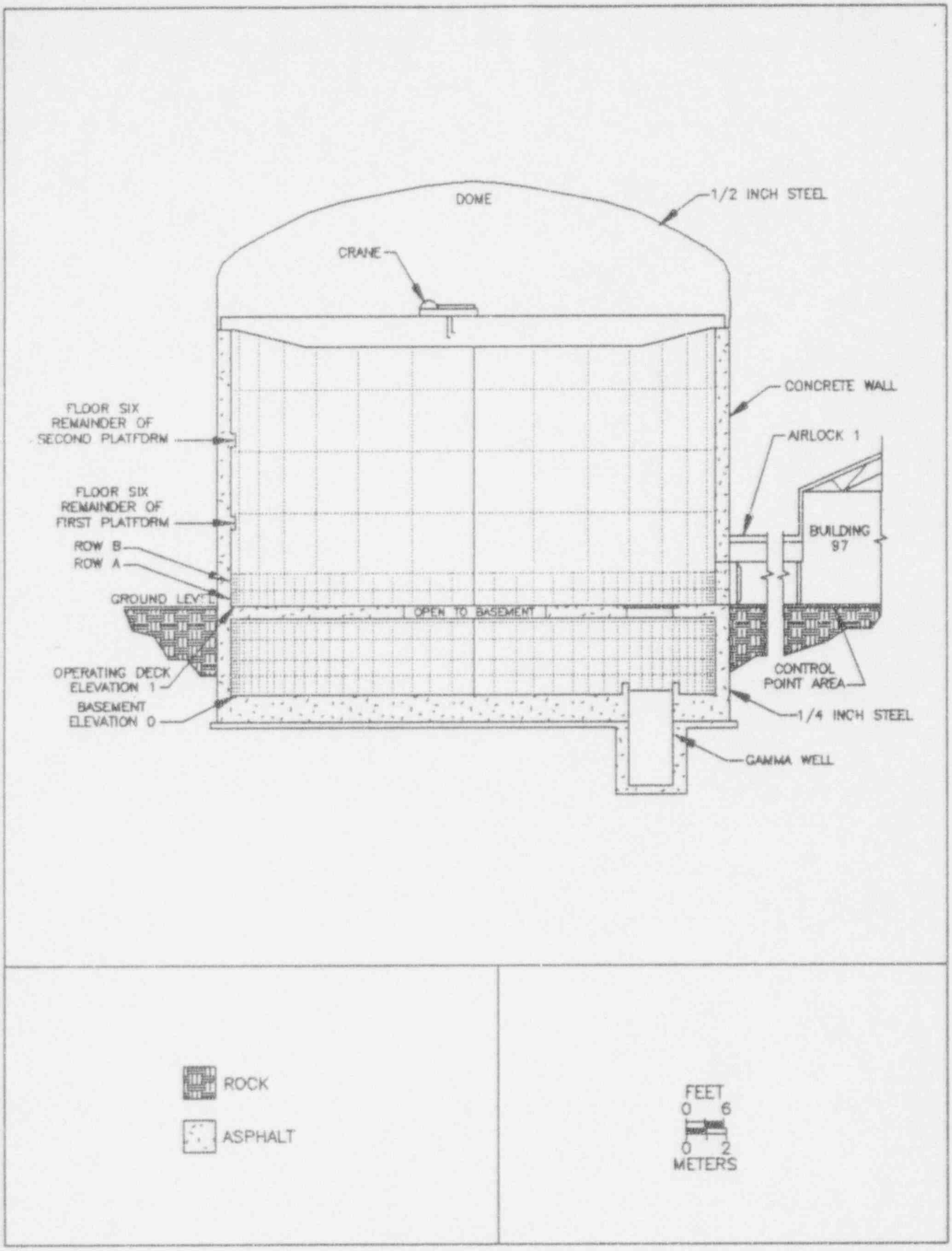

FIGURE 4: Cross Section View of Reactor, Building 100

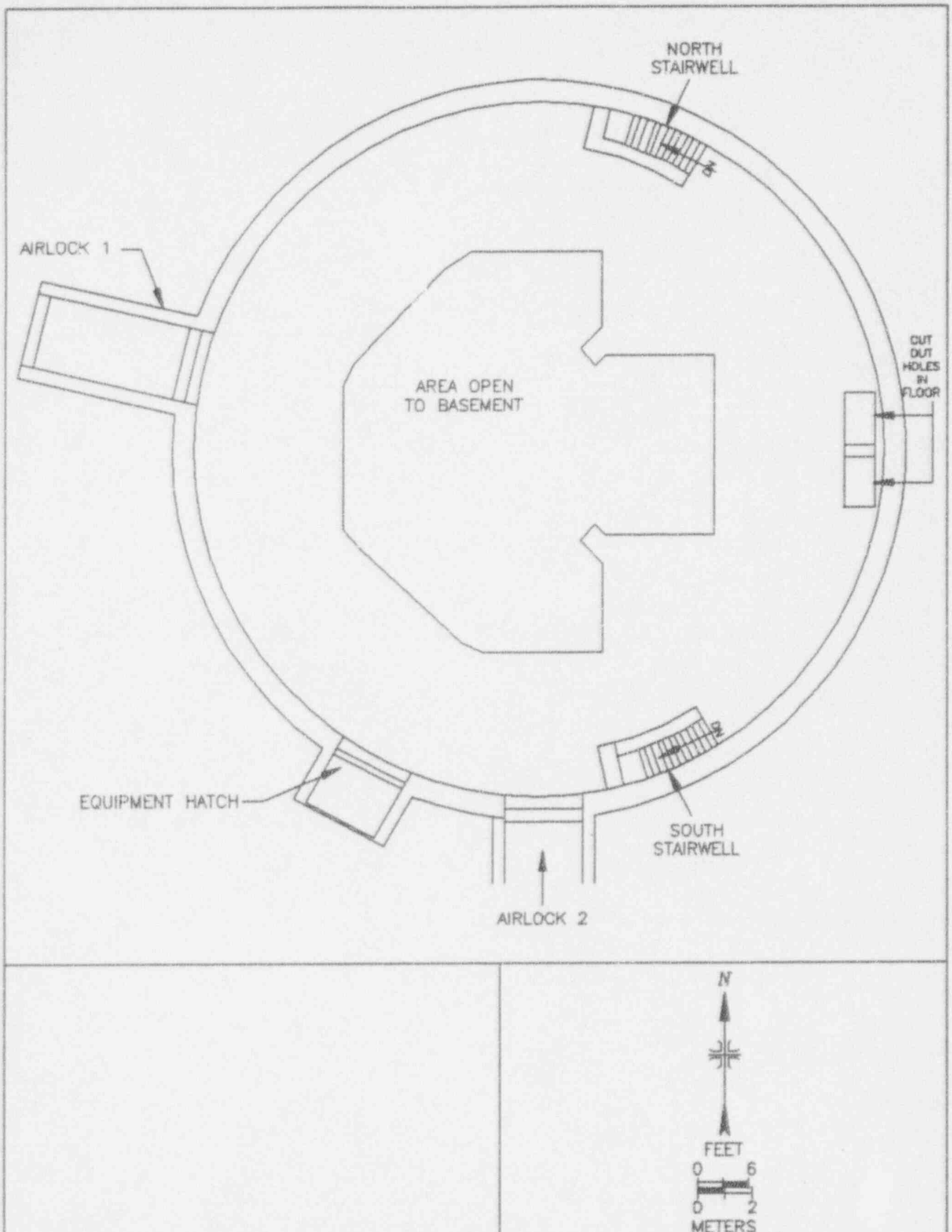

FIGURE 5: Plot Plan of Building 100, Operating Deck Floor

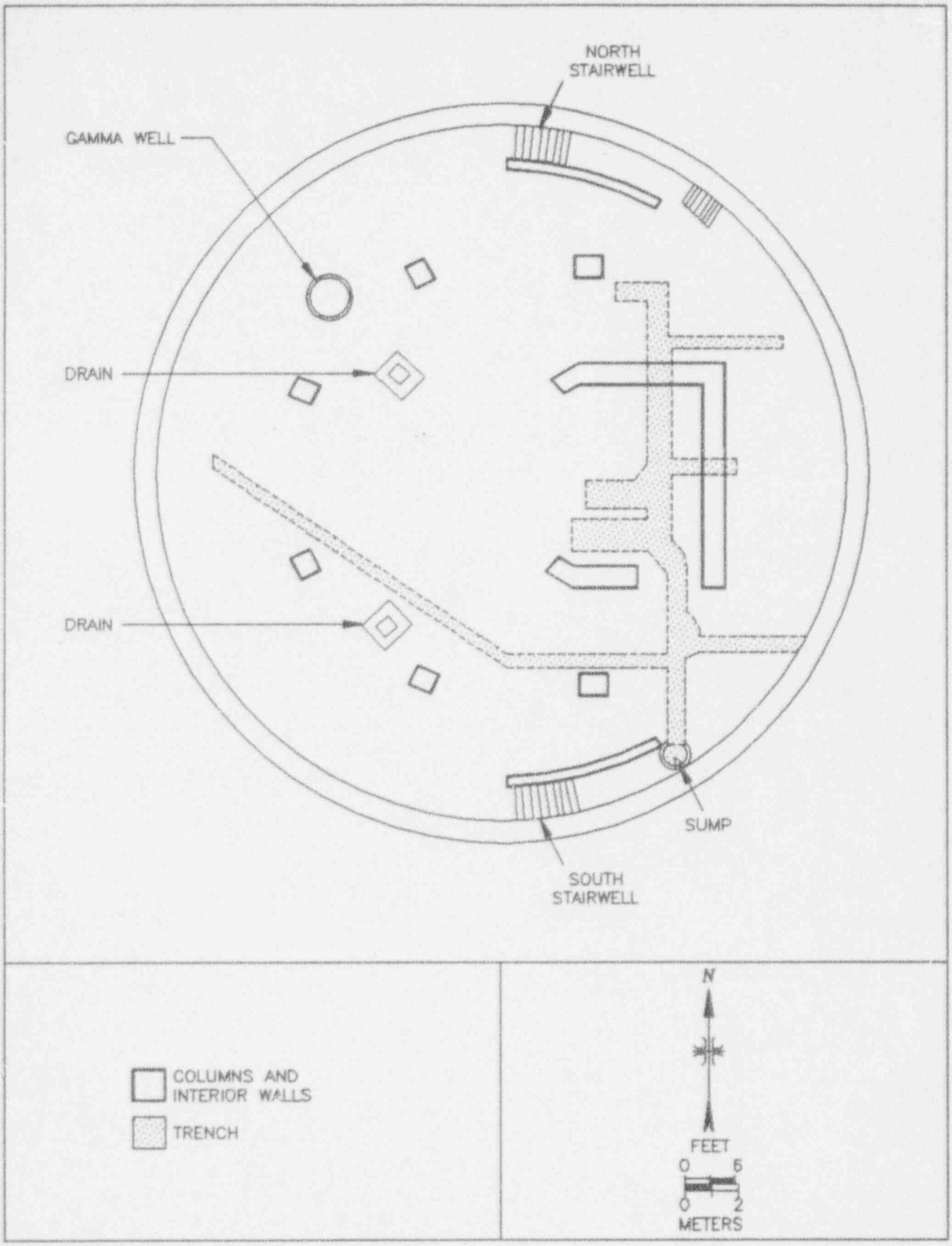

FIGURE 6: Plot Plan of Building 100, Basement Level Floor

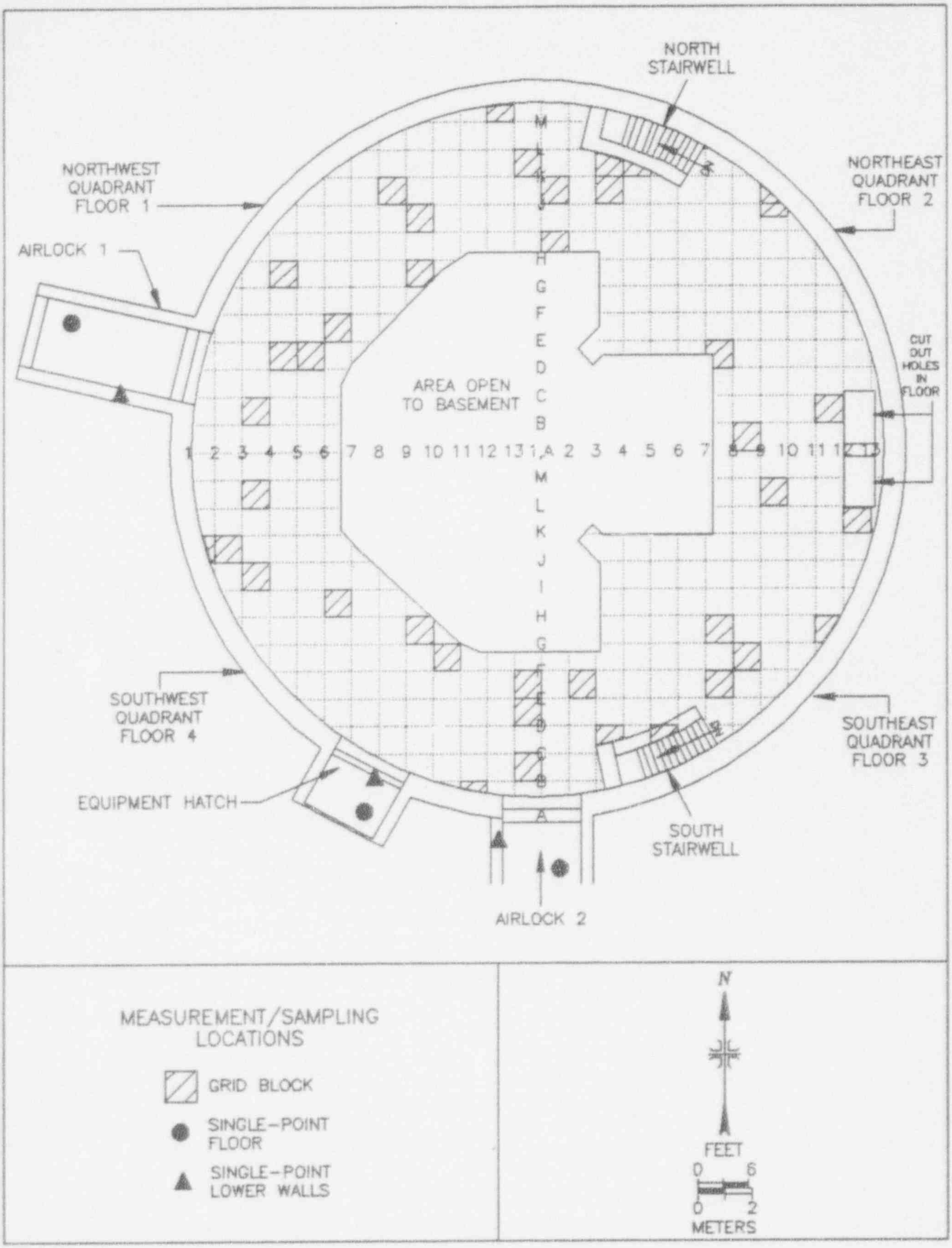

FIGURE 7: Building 100, Operating Deck Floor – Reference Grid and Measurement and Sampling Locations

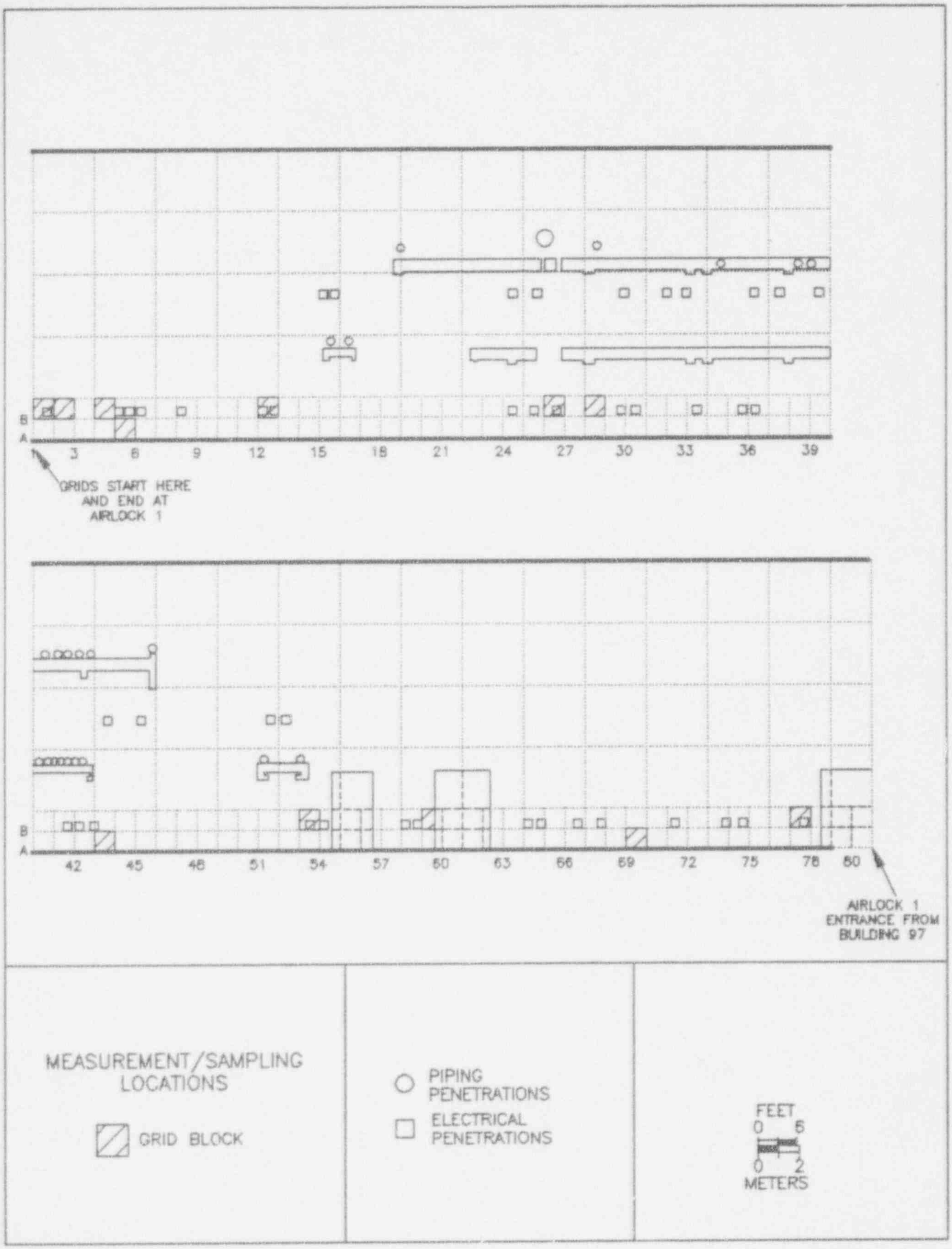

FIGURE 8: Building 100, Operating Deck Wall — Reference Grid and Measurement and Sampling Locations

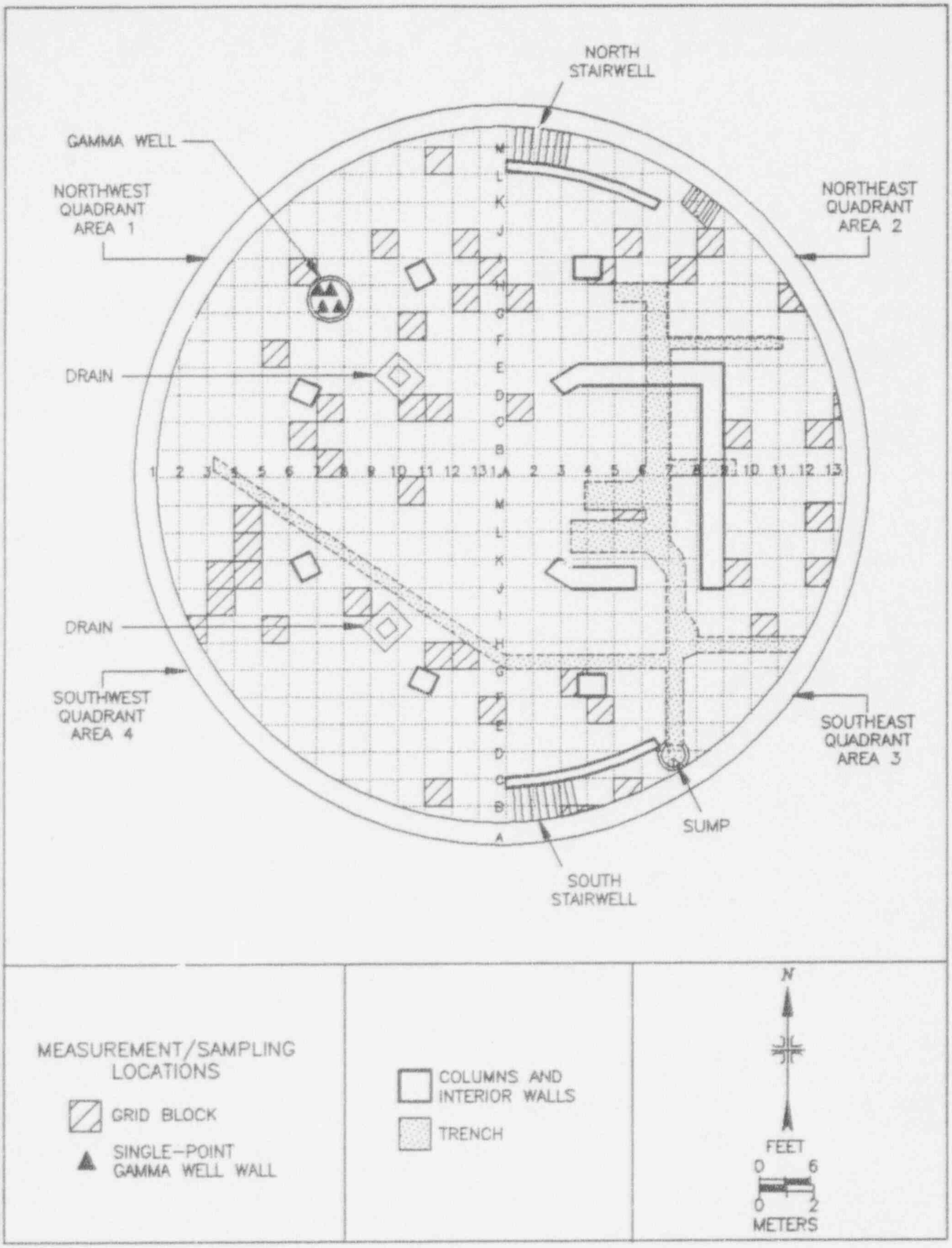

FIGURE 9: Building 100, Basement Level Floor – Reference Grid and Measurement and Sampling Locations

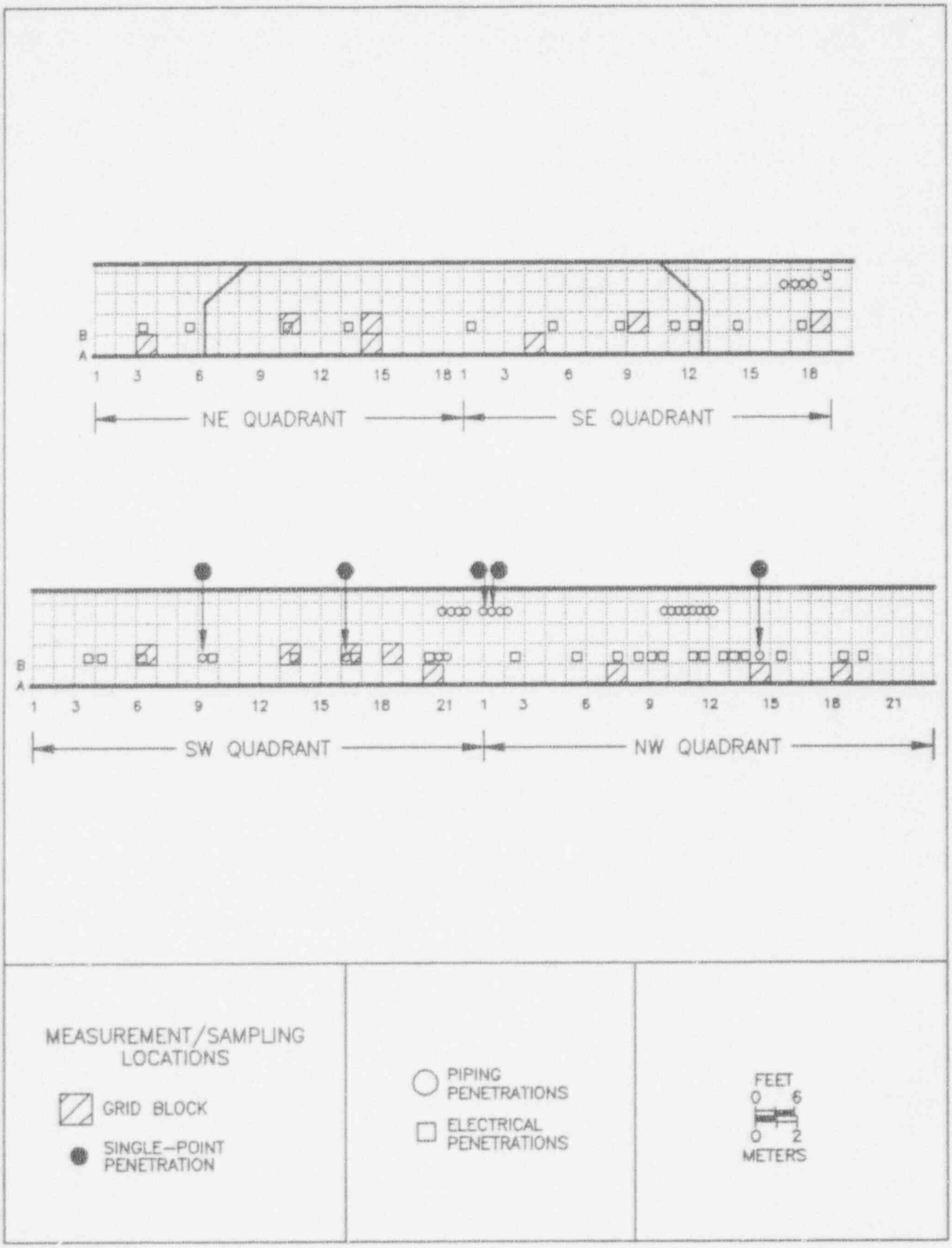

FIGURE 10: Building 100, Basement Level Walls – Reference Grid and Measurement and Sampling Locations

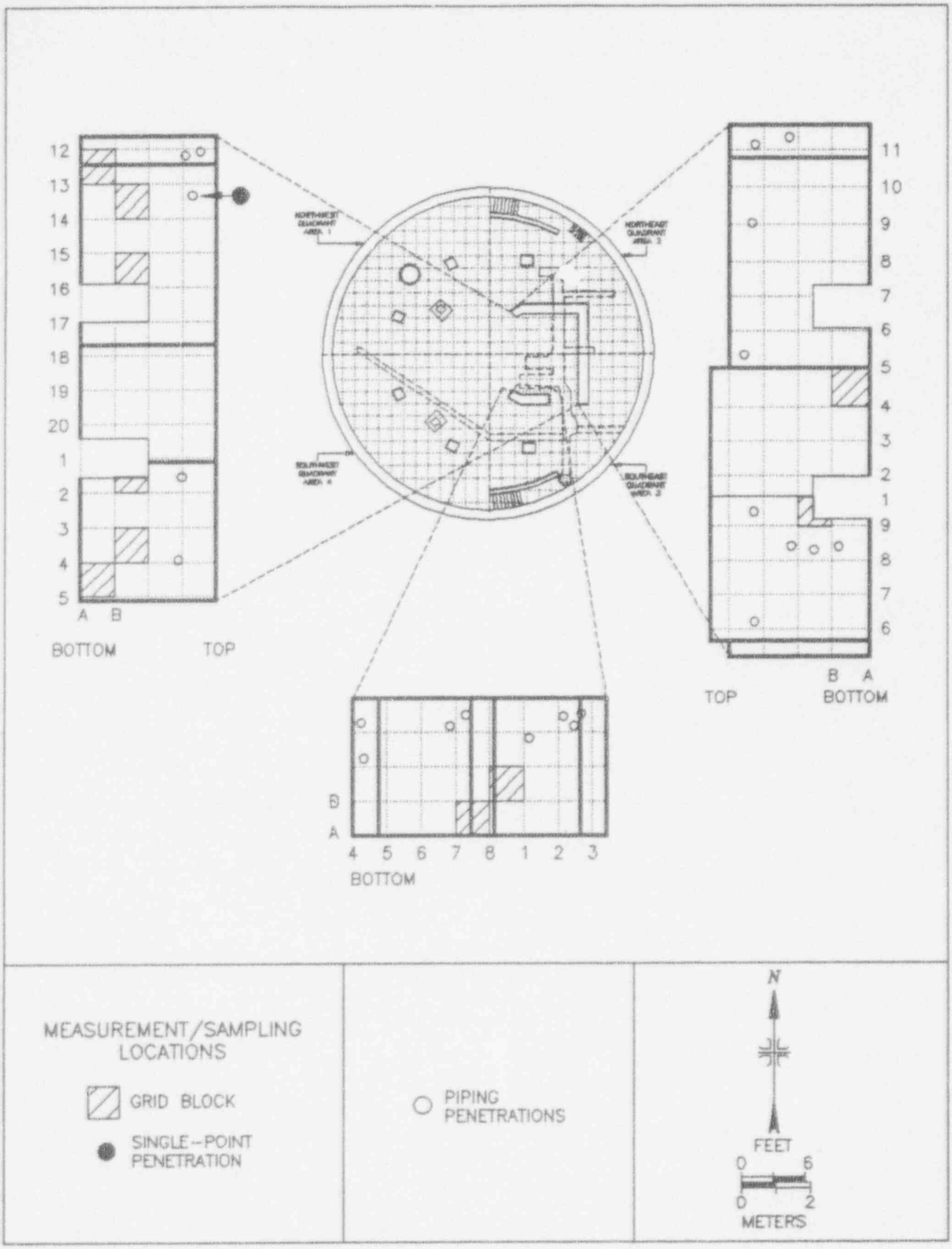

FIGURE 11: Building 100, Basement Inner Walls — Reference Grid and Measurement and Sampling Locations

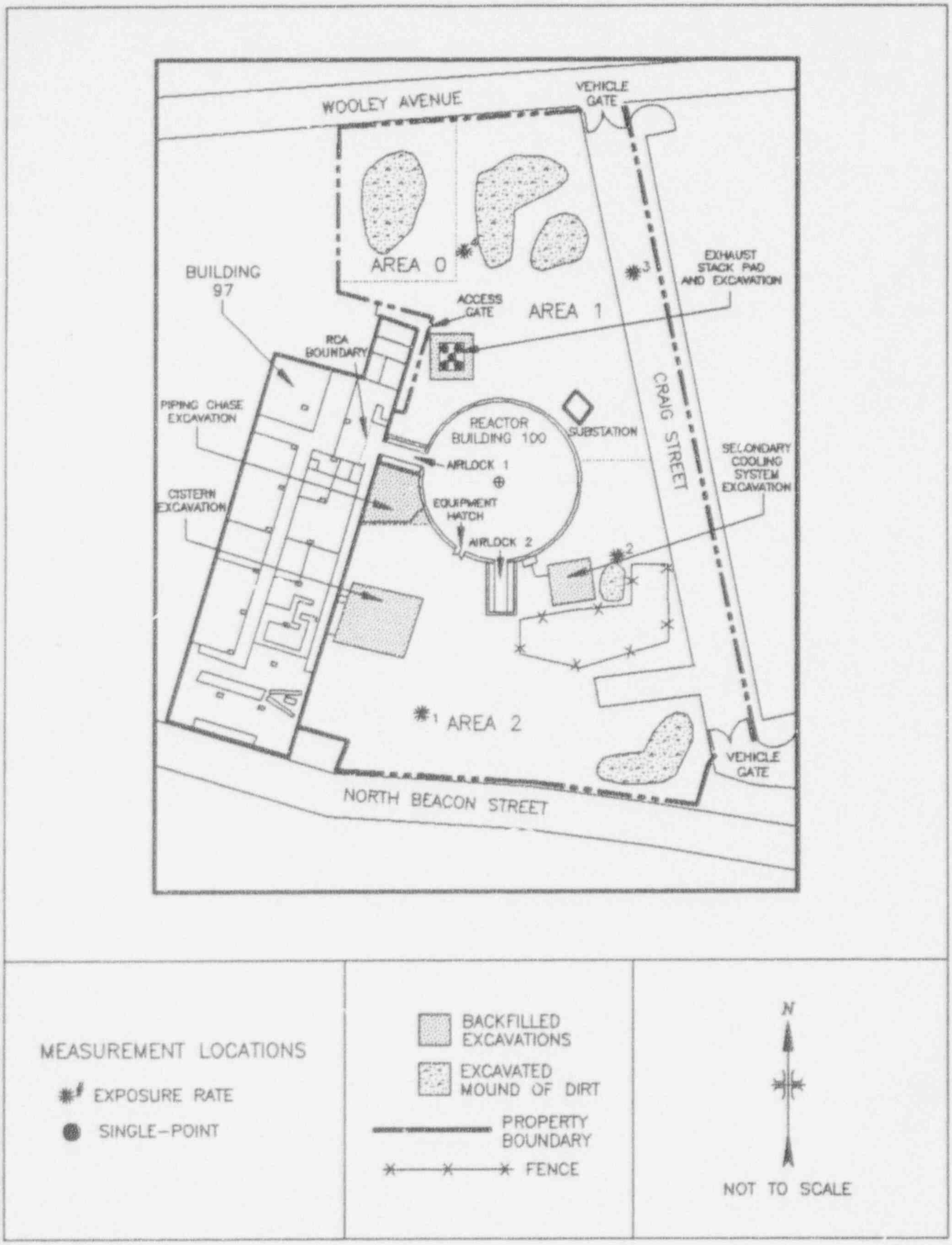

FIGURE 12: Army Materials Technology Laboratory, Reactor Yard – Measurement Locations

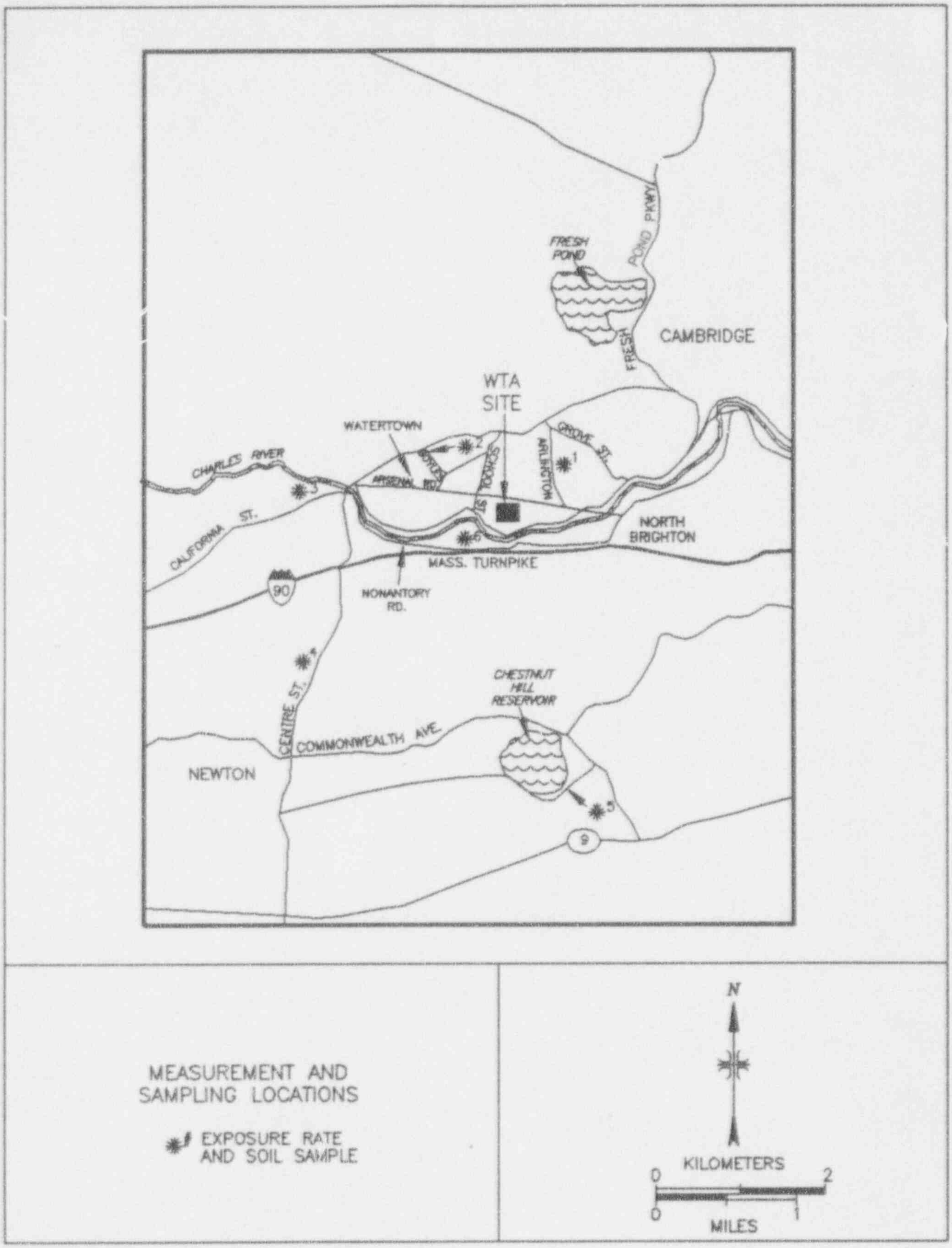

FIGURE 13: Watertown, Massachusetts Area — Background Measurement and Sampling Locations

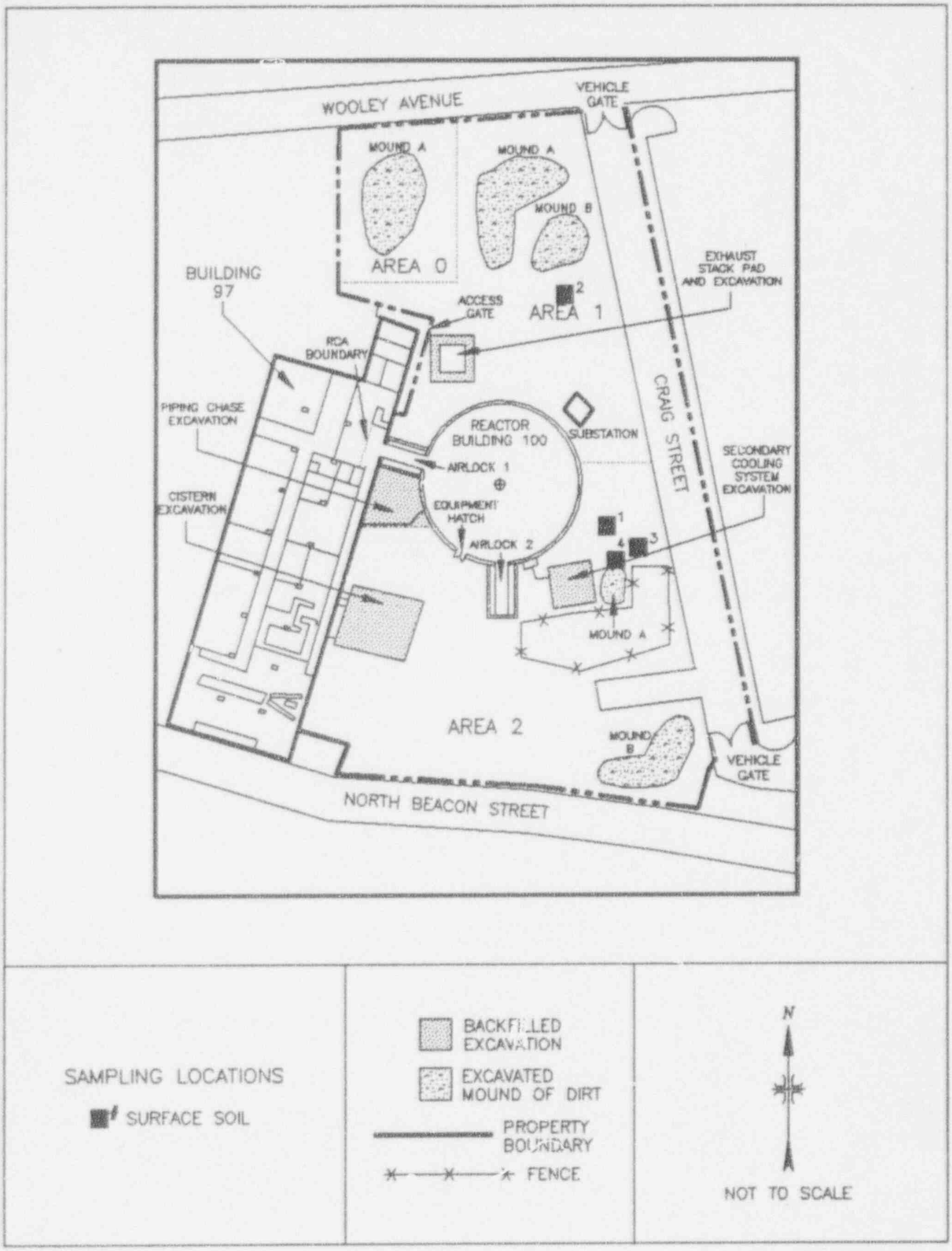

FIGURE 14: Army Materials Technology Laboratory, Reactor Yard — Soil Sampling Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS
ARMY MATERIALS TESTING LABORATORY
WATERTOWN ARSENAL
WATERTOWN, MASSACHUSETTS

Location[a]	Number of Measurement Locations		Range of Total Activity (dpm/100 cm²)				Range of Removable Activity (dpm/100 cm²)	
			Single Measurements		Grid Block Average			
	Single-pt.	Grid Blocks	Alpha	Beta	Alpha	Beta	Alpha	Beta
Operation Deck								
Floor	NA	38	<83	<1400	<83	<1400	<12	<17
Lower Walls	NA	12	<83	<1400	<83	<1400	<12	<17
Airlock Floors	2	NA	<83	<1400	NA	NA	<12	<17
Airlock Walls	2	NA	<83	<1400	NA	NA	<12	<17
Equip. Hatch. Fl.	1	NA	<83	<1400	NA	NA	<12	<17
Equip. Hatch LW	1	NA	<83	<1400	NA	NA	<12	<17
Basement Deck								
Floor	NA	45	<83	<1400 - 1500	<83	<1400	<12	<17
Lower Walls	NA	25	80	<1500	<73	<1500	<12	<17
Wall Penetrations	6	NA	<73	<1500	NA	NA	<12	<17
Gamma Well	4	NA	NA	<1400	NA	NA	<12	<17
Reactor Yard								
Exhaust Stack Pad	5	NA	<73	<1500	NA	NA	<12	<17

[a]Refer to Figures 7 through 12
[b]NA: Not applicable

TABLE 2

BACKGROUND EXPOSURE RATES AND RADIONUCLIDE CONCENTRATIONS IN SOIL
ARMY MATERIALS TESTING LABORATORY
WATERTOWN ARSENAL
WATERTOWN, MASSACHUSETTS

Location[a]	Exposure Rate (μR/h) at 1 m	Radionuclide Concentration (pCi/g)					
		Co-60	Cs-137	Eu-152	Eu-154	U-235	U-238
Arlington St. Park	11	<0.1	0.1±0.1[b]	<0.3	<0.5	0.1±0.1	0.6±1.1
E. Jr. High at Boyleston St.	10	0.1±0.1	0.2±0.1	<0.2	<0.3	0.1±0.1	0.6±1.1
California St.	11	<0.1	0.4±0.1	<0.2	<0.3	0.1±0.1	1.2±0.8
Sacred Heart School	10	<0.1	0.7±0.2	<0.2	<0.5	0.1±0.1	0.4±1.3
Cleveland Circle	10	<0.1	0.1±0.1	<0.2	<0.4	0.1±0.1	1.1±1.4
Nanatury Road Park	10	<0.2	0.4±0.1	<0.3	<0.4	0.1±0.1	1.1±0.9

[a]Refer to Figure 13
[b]Uncertainties represent the 95% confidence level, based only on counting statistics.

TABLE 3

EXPOSURE RATES
ARMY MATERIAL TESTING LABORATORY
WATERTOWN ARSENAL
WATERTOWN, MASSACHUSETTS

Location[a]	Exposure Rate at 1 m (μR/h)
Reactor Yard #1	11
Reactor Yard #2	11
Reactor Yard #3	11
Reactor Yard #4	12

[a]Refer to Figure 12

TABLE 4

RADIONUCLIDE CONCENTRATIONS IN SOIL
ARMY MATERIALS TESTING LABORATORY
WATERTOWN ARSENAL
WATERTOWN, MASSACHUSETTS

Location[a]	Radionuclide Concentration (pCi/g)					
	Co-60	Cs-137	Eu-152	Eu-154	U-235	U-238
Reactor Yard #1[c]	<0.1	<0.1	<0.2	<0.3	0.1 ± 0.1[b]	1.8 ± 0.7
Reactor Yard #2[c]	<0.1	0.1 ± 0.1	<0.2	<0.3	0.2 ± 0.1	1.2 ± 1.1
Reactor Yard #3[c]	<0.1	<0.1	<0.2	<0.3	0.1 ± 0.1	1.1 ± 0.5
Reactor Yard #4[c]	<0.1	<0.1	<0.1	<0.3	0.1 ± 0.1	0.9 ± 0.9
Area 2, Cistern 242 Excavation[d]	<0.1	0.1 ± 0.1	<0.2	<0.4	0.1 ± 0.1	0.6 ± 0.8
Area 2, Cistern 242 Excavation[d]	0.1 ± 0.1	0.1 ± 0.1	<0.2	<0.3	0.1 ± 0.1	1.3 ± 0.8
Area 2, Cistern 242 Excavation[d]	<0.1	<0.1	<0.2	<0.3	0.2 ± 0.1	0.9 ± 1.5
Area 2, Cistern 242 Excavation[d]	<0.1	0.1 ± 0.1	<0.2	<0.3	0.2 ± 0.1	0.8 ± 1.0
Area 2, Cistern 242 Excavation[d]	0.1 ± 0.1	<0.1	<0.2	<0.4	0.1 ± 0.1	1.0 ± 1.1
Area 2, Pipe Chase Excavation[e]	<0.1	0.2 ± 0.1	<0.2	<0.2	0.1 ± 0.1	1.0 ± 1.1
Area 2, Pipe Chase Excavation[e]	<0.1	0.3 ± 0.1	<0.3	<0.3	0.1 ± 0.8	1.0 ± 1.1
Area 1, Mound A[e]	<0.1	0.1 ± 0.1	<0.2	<0.3	0.2 ± 0.1	1.7 ± 1.[illegible]
Area 1, Mound B[e]	<0.1	<0.1	<0.3	<0.3	0.2 ± 0.8	1.6 ± 1.3
Area 1, Mound B[e]	0.1 ± 0.1	0.1 ± 0.1	<0.2	<0.3	0.2 ± 0.1	<1.0
Area 2, Mound B[e]	<0.1	0.2 ± 0.1	<0.2	<0.3	0.1 ± 0.1	2.0 ± 1.4
Area 2, Mound B[e]	<0.1	0.2 ± 0.1	<0.2	<0.5	0.1 ± 0.1	1.7 ± 1.6
Area 2, Secondary Cooling System Excavation[e]	<0.1	<0.1	<0.2	<0.3	0.2 ± 0.1	1.4 ± 1.3
Area 2, Secondary Cooling System Excavation[e]	<0.1	<0.1	<0.2	<0.3	0.2 ± 0.1	0.5 ± 1.1

[a]Refer to Figure 14
[b]Uncertainties represent the 95% confidence level, based only on counting statistics.
[c]ESSAP collected sample
[d]NRC collected sample
[e]SEG collected sample

REFERENCES

1. Scientific Ecology Group, Inc., "Army Materials Technology Laboratory, Research Reactor Decommissioning, Final Survey Report, Revision 1" May 1993.

2. ORISE, letter from T. J. Vitkus to T. Dragoun, U.S. Nuclear Regulatory Commission, "Proposed Survey Plan for the Army Materials Technology Laboratory, Research Reactor," April 20, 1993.

3. ORISE, letter from T. J. Vitkus to T. Dragoun, U.S. Nuclear Regulatory Commission, "Document Review of the Army Materials Technology Laboratory, Research Reactor Decommissioning Final Survey Report," April 16, 1993.

4. U.S. Nuclear Regulatory Commission, "Guidance and Discussion of Requirement for an Application to Terminate a Non-Power Reactor Facility Operating License," Rev. 1, September 1984.

APPENDIX A
MAJOR INSTRUMENTATION

APPENDIX A
MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the authors or their employers.

DIRECT RADIATION MEASUREMENT

Instruments

Eberline Pulse Ratemeter
Model PRM-6
(Eberline, Santa Fe, NM)

Eberline "Rascal" Ratemeter-Scaler
Model PRS-1
(Eberline, Santa Fe, NM)

Ludlum Floor Monitor
Model 239-1
(Ludlum Measurements, Inc., Sweetwater, TX)

Ludlum Ratemeter-Scaler
Model 2221
(Ludlum Measurements, Inc., Sweetwater, TX)

Detectors

Eberline GM Detector
Model HP-260
Effective Area, 15.5 cm^2
(Eberline, Santa Fe, NM)

Eberline ZnS Scintillation Detector
Model AC-3-7
Effective Area, 59 cm^2
(Eberline, Santa Fe, NM)

Ludlum Gas Proportional Detector
Model 43-37
Effective Area, 550 cm^2
(Ludlum Measurements, Inc., Sweetwater, TX)

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

Victoreen NaI Scintillation Detector
Model 489-55
3.2 cm x 3.8 cm Crystal
(Victoreen, Cleveland, OH)

LABORATORY ANALYTICAL INSTRUMENTATION

High Purity Extended Range Intrinsic Detectors
Model No: ERVDS30-25195
(Tennelec, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, TN) and
Multichannel Analyzer
3100 Vax Workstation
(Canberra, Meriden, CT)

High-Purity Germanium Detector
Model GMX-23195-S, 23% Eff.
(EG&G ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and
Multichannel Analyzer
3100 Vax Workstation
(Canberra, Meriden, CT)

Low Background Gas Proportional Counter
Model LB-5110
(Tennelec, Oak Ridge, TN)

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

SURVEY PROCEDURES

Surface Scans

Surface scans were performed by passing the probes slowly over the surface; the distance between the probe and the surface was maintained at a minimum - nominally about 1 cm. A large surface area, gas proportional floor monitor was used to scan the floors and walls of the surveyed areas. Other surfaces were scanned using small area (15.5 cm^2 or 59 cm^2) hand-held detectors. Identification of elevated levels was based on increases in the audible signal from the recording and/or indicating instrument. Combinations of detectors and instruments used for the scans were:

Alpha	-	gas proportional detector with ratemeter-scaler
	-	ZnS scintillation detector with ratemeter-scaler
Beta	-	gas proportional detector with ratemeter-scaler
	-	pancake GM detector with ratemeter-scaler
Gamma	-	NaI scintillation detector with ratemeter

Surface Activity Measurements

Measurements of total alpha and total beta activity levels were performed using ZnS scintillation and GM detectors with ratemeters-scalers.

Count rates (cpm) integrated over 1 minute in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the net rate by the 4π efficiency and correcting for the active area of the detector. The alpha activity background countrates for the ZnS scintillation detectors were 1 cpm for each detector. Alpha efficiency factors ranged from 0.15 - 0.17 for the ZnS scintillation detectors. The beta activity background count rates for the GM detectors averaged 49 cpm. Beta efficiency factors were 0.16 for the GM detectors. The effective window for the ZnS scintillation and GM detectors were 59 cm^2 and 15.5 cm^2, respectively.

Removable Activity Measurements

Removable activity levels were determined using numbered filter paper disks, 47 mm in diameter. Moderate pressure was applied to the smear and approximately 100 cm^2 of the surface was wiped. Smears were placed in labeled envelopes with the location and other pertinent information recorded.

Exposure Rate Measurements

Measurements of gamma exposure rates were performed using a pressurized ionization chamber (PIC).

Soil Sampling

Approximately 1 kg of soil was collected at each sample location. Collected samples were placed in a plastic bag, sealed, and labeled in accordance with ESSAP survey procedures.

ANALYTICAL PROCEDURES

Radiological Analyses

Removable Activity

Smears were counted on a low background gas proportional system for gross alpha and gross beta activity.

Gamma Spectrometry

Soil Samples

Soil samples were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in 0.5-liter Marinelli beakers or other appropriate containers. The quantity placed in each beaker was chosen to reproduce the calibrated counting geometry and ranged from 500 to 1100 g of material. Net material weights were determined and the samples counted using intrinsic germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. Energy peaks used for determination of radionuclides of concern were:

Co-60	1.173 MeV
Cs-137	0.662 MeV
Eu-152	0.344 MeV
Eu-154	0.723 MeV

U-235	0.143 MeV (or 0.186 MeV)
U-238	0.063 MeV or 0.093 MeV from Th-234* (or 1.001 MeV from Pa-234 m)*

*Secular equilibrium assumed.

Spectra were also reviewed for other identifiable photopeaks.

UNCERTAINTIES AND DETECTION LIMITS

The uncertainties associated with the analytical data presented in the tables of this report represent the 95% confidence level for that data. These uncertainties were calculated based on both the gross sample count levels and the associated background count levels. When the net sample count was less than 2.71 + 4.66 times the statistical deviation of the background count, the sample concentration was reported as less than the detection limit of the measurement procedures. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument. Additional uncertainties, associated with sampling and measurement procedures, have not been propagated into the data presented in this report.

CALIBRATION AND QUALITY ASSURANCE

Analytical and field survey activities were conducted in accordance with procedures from the following documents of the Environmental Survey and Site Assessment Program:

- Survey Procedures Manual Revision 7 (May 1992)
- Laboratory Procedures Manual Revision 7 (April 1992)
- Quality Assurance Manual Revision 5 (May 1992)

The procedures contained in these manuals were developed to meet the requirements of DOE Order 5700.6C and ASME NQA-1 for Quality Assurance and contain measures to assess processes during their performance.

Calibration of all field and laboratory instrumentation was based on standards/sources, traceable to NIST.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations,
- Participation in EPA and EML Quality Assurance Programs,
- Training and certification of all individuals performing procedures, and
- Periodic internal and external audits.

APPENDIX C

REGULATORY GUIDE 1.86
TERMINATION OF OPERATING LICENSES
FOR NUCLEAR REACTORS

U.S. ATOMIC ENERGY COMMISSION

June 1974

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

A. INTRODUCTION

Section 50.51, "Duration of license, renewal," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each license to operate a production and utilization facility be issued for a specified duration. Upon expiration of the specified period, the license may be either renewed or terminated by the Commission. Section 50.82, "Applications for termination of licenses," specifies the requirements that must be satisfied to terminate an operating license, including the requirement that the dismantlement of the facility and disposal of the component parts not be inimical to the common defense and security or to the health and safety of the public. This guide describes methods and procedures considered acceptable by the Regulatory staff for the termination of operating licenses for nuclear reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

When a licensee decides to terminate his nuclear reactor operating license, he may, as a first step in the process, request that his operating license be amended to restrict him to possess but not operate the facility. The advantage to the licensee of converting to such a possession-only license is reduced surveillance requirements in that periodic surveillance of equipment important to the safety of reactor operation is no longer required. Once this possession-only license is issued, reactor operation is not permitted. Other activities related to cessation of operations such as unloading fuel from the reactor and placing it in storage (either onsite of offsite) may be continued.

A licensee having a possession-only license must retain, with the Part 50 license, authorization for special nuclear material (10 CFR Part 70, "Special Nuclear Material"), byproduct material (10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material"), and source material (10 CFR Part 40, "Licensing of Source Material"), until the fuel, radioactive components, and sources are removed from the facility. Appropriate administrative controls and facility requirements are imposed by the Part 50 license and the technical specifications to assure that proper surveillance is performed and that the reactor facility is maintained in a safe condition and not operated.

A possession-only license permits various options and procedures for decommissioning, such as mothballing, entombment, or dismantling. The requirements imposed depend on the option selected.

Section 50.82 provides that the licensee may dismantle and dispose of the component parts of a nuclear reactor in accordance with existing regulations. For research reactors and critical facilities, this has usually meant the disassembly of a reactor and its shipment offsite, sometimes to another appropriately licensed organization for further use. The site from which a reactor has been removed must be decontaminated, as necessary, and inspected by the Commission to determine whether unrestricted access can be approved. In the case of nuclear power reactors, dismantling has usually been accomplished by shipping fuel offsite, making the reactor inoperable, and disposing of some of the radioactive components.

Radioactive components may be either shipped offsite for burial at an authorized burial ground or secured

on the site. Those radioactive materials remaining on the site must be isolated from the public by physical barriers or other means to prevent public access to hazardous levels of radiation. Surveillance is necessary to assure the long term integrity of the barriers. The amount of surveillance required depends upon (1) the potential hazard to the health and safety of the public from radioactive material remaining on the site and (2) the integrity of the physical barriers. Before areas may be released for unrestricted use, they must have been decontaminated or the radioactivity must have decayed to less than prescribed limits (Table I).

The hazard associated with the retired facility is evaluated by considering the amount and type of remaining contamination, the degree of confinement of the remaining radioactive materials, the physical security provided by the confinement, the susceptibility to release of radiation as a result of natural phenomena, and the duration of required surveillance.

C. REGULATORY POSITION

1. APPLICATION FOR A LICENSE TO POSSESS BUT NOT OPERATE (POSSESSION-ONLY LICENSE)

A request to amend an operating license to a possession-only license should be made to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545. The request should include the following information:

a. A description of the current status of the facility.

b. A description of measures that will be taken to prevent criticality or reactivity changes and to minimize releases of radioactivity from the facility.

c. Any proposed changes to the technical specifications that reflect the possession-only facility status and the necessary disassembly/retirement activities to be performed.

d. A safety analysis of both the activities to be accomplished and the proposed changes to the technical specifications.

e. An inventory of activated materials and their location in the facility.

2. ALTERNATIVES FOR REACTOR RETIREMENT

Four alternatives for retirement of nuclear reactor facilities are considered acceptable by the Regulatory staff. These are:

a. **Mothballing.** Mothballing of a nuclear reactor facility consists of putting the facility in a state of protective storage. In general, the facility may be left intact except that all fuel assemblies and the radioactive fluids and waste should be removed from the site. Adequate radiation monitoring, environmental surveillance, and appropriate security procedures should be established under a possession-only license to ensure that the health and safety of the public is not endangered.

b. **In-Place Entombment.** In-place entombment consists of sealing all the remaining highly radioactive or contaminated components (e.g., the pressure vessel and reactor internals) within a structure integral with the biological shield after having all fuel assemblies, radioactive fluids and wastes, and certain selected components shipped offsite. The structure should provide integrity over the period of time in which significant quantities (greater than Table I levels) of radioactivity remain with the material in the entombment. An appropriate and continuing surveillance program should be established under a possession-only license.

c. **Removal of Radioactive Components and Dismantling.** All fuel assemblies, radioactive fluids and waste, and other materials having activities above accepted unrestricted activity levels (Table I) should be removed from the site. The facility owner may then have unrestricted use of the site with no requirement for a license. If the facility owner so desires, the remainder of the reactor facility may be dismantled and all vestiges removed and disposed of.

d. **Conversion to a New Nuclear System or a Fossil Fuel System.** This alternative, which applies only to nuclear power plants, utilizes the existing turbine system with a new steam supply system. The original nuclear steam supply system should be separated from the electric generating system and disposed of in accordance with one of the previous three retirement alternatives.

3. SURVEILLANCE AND SECURITY FOR THE RETIREMENT ALTERNATIVES WHOSE FINAL STATUS REQUIRES A POSSESSION-ONLY LICENSE

A facility which has been licensed under a possession-only license may contain a significant amount of radioactivity in the form of activated and contaminated hardware and structural materials. Surveillance and commensurate security should be provided to assure that the public health and safety are not endangered.

a. Physical security to prevent inadvertent exposure of personnel should be provided by multiple locked barriers. The presence of these barriers should make it extremely difficult for an unauthorized person to gain access to areas where radiation or contamination levels exceed those specified in Regulatory Position C.4. To prevent inadvertent exposure, radiation areas above 5 mR/hr, such as near the activated primary system of a power plant, should be appropriately marked and should not be accessible except by cutting of welded closures or the disassembly and removal of substantial structures

and/or shielding material. Means such as a remote-readout intrusion alarm system should be provided to indicate to designated personnel when a physical barrier is penetrated. Security personnel that provide access control to the facility may be used instead of the physical barriers and the intrusion alarm systems.

b. The physical barriers to unauthorized entrance into the facility, e.g., fences, buildings, welded doors, and access openings, should be inspected at least quarterly to assure that these barriers have not deteriorated and that locks and locking apparatus are intact.

c. A facility radiation survey should be performed at least quarterly to verify that no radioactive material is escaping or being transported through the containment barriers in the facility. Sampling should be done along the most probable path by which radioactive material such as that stored in the inner containment regions could be transported to the outer regions of the facility and ultimately to the environs.

d. An environmental radiation survey should be performed at least semiannually to verify that no significant amounts of radiation have been released to the environment from the facility. Samples such as soil, vegetation, and water should be taken at locations for which statistical data has been established during reactor operations.

e. A site representative should be designated to be responsible for controlling authorized access into and movement within the facility.

f. Administrative procedures should be established for the notification and reporting of abnormal occurrences such as (1) the entrance of an unauthorized person or persons into the facility and (2) a significant change in the radiation or contamination levels in the facility or the offsite environment.

g. The following reports should be made:

(1) An annual report to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, describing the results of the environmental and facility radiation surveys, the status of the facility, and an evaluation of the performance of security and surveillance measures.

(2) An abnormal occurrence report to the Regulatory Operations Regional Office by telephone within 24 hours of discovery of an abnormal occurrence. The abnormal occurrence will also be reported in the annual report described in the preceding item.

h. Records or logs relative to the following items should be kept and retained until the license is terminated, after which they may be stored with other plant records:

(1) Environmental surveys,

(2) Facility radiation surveys,

(3) Inspections of the physical barriers, and

(4) Abnormal occurrences.

4. DECONTAMINATION FOR RELEASE FOR UNRESTRICTED USE

If it is desired to terminate a license and to eliminate any further surveillance requirements, the facility should be sufficiently decontaminated to prevent risk to the public health and safety. After the decontamination is satisfactorily accomplished and the site inspected by the Commission, the Commission may authorize the license to be terminated and the facility abandoned or released for unrestricted use. The licensee should perform the decontamination using the following guidelines:

a. The licensee should make a reasonable effort to eliminate residual contamination.

b. No covering should be applied to radioactive surfaces of equipment or structures by paint, plating, or other covering material until it is known that contamination levels (determined by a survey and documented) are below the limits specified in Table I. In addition, a reasonable effort should be made (and documented) to further minimize contamination prior to any such covering.

c. The radioactivity of the interior surfaces of pipes, drain lines, or ductwork should be determined by making measurements at all traps and other appropriate access points, provided 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 should be assumed to be contaminated in excess of the permissable radiation limits.

d. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated in excess of the limits specified. This may include, but is not limited to, special circumstances such as the transfer of premises to another licensed organization that will continue to work with radioactive materials. Requests for such authorization should provide:

(1) Detailed, specific information describing the premises, equipment, scrap, and radioactive contaminants and the nature, extent, and degree of residual surface contamination.

(2) A detailed health and safety analysis indicating that the residual amounts of materials on surface areas, together with other considerations such as the prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

e. Prior to release of the premises for unrestricted use, the licensee should make a comprehensive radiation survey establishing that contamination is within the limits specified in Table I. A survey report should be filed with the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, with a copy to the Director of the Regulatory Operations Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report should:

(1) Identify the premises;

(2) Show that reasonable effort has been made to reduce residual contamination to as low as practicable levels;

(3) Describe the scope of the survey and the general procedures followed; and

(4) State the finding of the survey in units specified in Table I.

After review of the report, the Commission may inspect the facilities to confirm the survey prior to granting approval for abandonment.

5. REACTOR RETIREMENT PROCEDURES

As indicated in Regulatory Position C.2, several alternatives are acceptable for reactor facility retirement. If minor disassembly or "mothballing" is planned, this could be done by the existing operating and maintenance procedures under the license in effect. Any planned actions involving an unreviewed safety question or a change in the technical specifications should be reviewed and approved in accordance with the requirements of 10 CFR §50.59.

If major structural changes to radioactive components of the facility are planned, such as removal of the pressure vessel or major components of the primary system, a dismantlement plan including the information required by §50.82 should be submitted to the Commission. A dismantlement plan should be submitted for all the alternatives of Regulatory Position C.2 except mothballing. However, minor disassembly activities may still be performed in the absence of such a plan, provided they are permitted by existing operating and maintenance procedures. A dismantlement plan should include the following:

a. A description of the ultimate status of the facility

b. A description of the dismantling activities and the precautions to be taken.

c. A safety analysis of the dismantling activities including any effluents which may be released.

d. A safety analysis of the facility in its ultimate status.

Upon satisfactory review and approval of the dismantling plan, a dismantling order is issued by the Commission in accordance with §50.82. When dismantling is completed and the Commission has been notified by letter, the appropriate Regulatory Operations Regional Office inspects the facility and verifies completion in accordance with the dismantlement plan. If residual radiation levels do not exceed the values in Table I, the Commission may terminate the license. If these levels are exceeded, the licensee retains the possession-only license under which the dismantling activities have been conducted or, as an alternative, may make application to the State (if an Agreement State) for a byproduct materials license.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
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	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/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.