CONFIRMATORY SURVEY OF THE BOELTER REACTOR FACILITY UNIVERSITY OF CALIFORNIA AT LOS ANGELES, CALIFORNIA

[DOCKET 50-142]

A. J. ANSARI AND J. L. PAYNE

Prepared for
U. S. Nuclear Regulatory Commission
Region V Office

ORISE

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

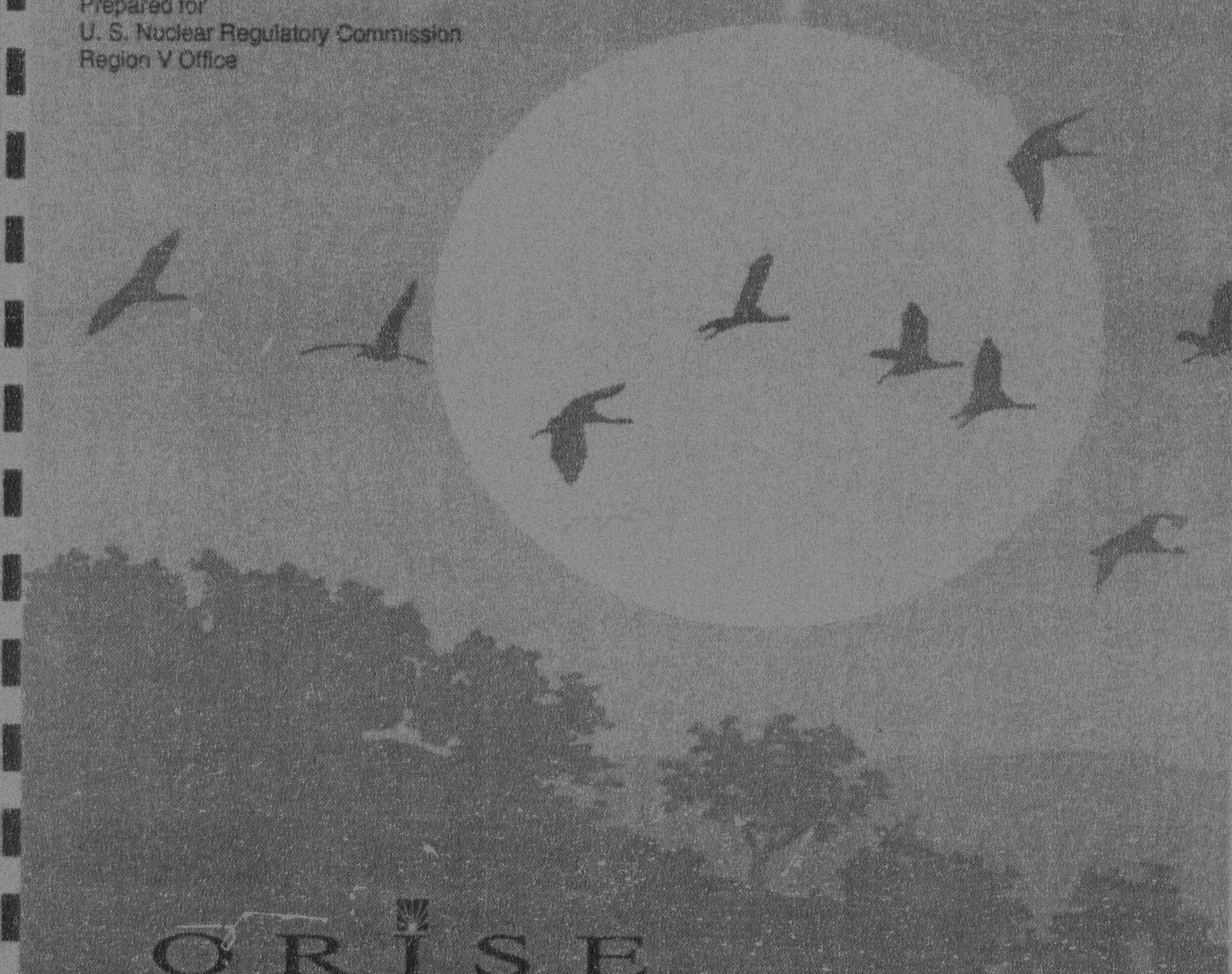
Environmental Survey and Site Assessment Program
Energy/Environment Systems Division

CONFIRMATORY SURVEY OF THE BOELTER REACTOR FACILITY UNIVERSITY OF CALIFORNIA AT LOS ANGELES, CALIFORNIA

[DOCKET 50-142]

A. J. ANSARI AND J. L. PAYNE

Prepared for
U. S. Nuclear Regulatory Commission
Region V Office



ORISE

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

Environmental Survey and Site Assessment Program
Energy/Environment Systems Division

The **Oak Ridge Institute for Science and Education** (ORISE) was established by the U.S. Department of Energy to undertake national and international programs in science and engineering education, training and management systems, energy and environment systems, and medical sciences. ORISE and its programs are operated by Oak Ridge Associated Universities (ORAU) through a management and operating contract with the U.S. Department of Energy. Established in 1946, ORAU is a consortium of 65 colleges and universities.

NOTICES

CONFIRMATORY SURVEY
OF THE
BOELTER REACTOR FACILITY
UNIVERSITY OF CALIFORNIA AT
LOS ANGELES, CALIFORNIA

Prepared by

A. J. Ansari and J. L. Payne

Environmental Survey and Site Assessment Program
Energy/Environment System Division
Oak Ridge Institute for Science and Education
Oak Ridge, Tennessee 37831-0117

Prepared for

U.S. Nuclear Regulatory Commission
Region V Office

August 1993

FINAL REPORT

This report is based on work performed under an Interagency Agreement (NRC Fin. No. A-9093) between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy. Oak Ridge Institute for Science and Education performs complementary work under contract number DE-AC05-76OR00033 with the U.S. Department of Energy.

CONFIRMATORY SURVEY
OF THE
BOELTER REACTOR FACILITY
UNIVERSITY OF CALIFORNIA AT
LOS ANGELES, CALIFORNIA

Prepared by: ______________________ Date: 8/24/93
A. J. Ansari, Project Leader
Environmental Survey and Site Assessment Program

Reviewed by: ______________________ Date: 8/24/93
M. J. Laudeman, Radiochemist/Laboratory Supervisor
Environmental Survey and Site Assessment Program

Reviewed by: ______________________ Date: 8/24/93
M. R. Landis, Project Manager
Environmental Survey and Site Assessment Program

Reviewed by: ______________________ Date: 8/24/93
A. T. Payne, Quality Assurance Officer
Environmental Survey and Site Assessment Program

Reviewed by: ______________________ Date: 8/24/93
J. D. Berger, Program Director
Environmental Survey and Site Assessment Program

ACKNOWLEDGEMENTS

The authors would like to acknowledge the significant contributions of the following staff members:

LABORATORY STAFF

R. D. Condra
J. S. Cox
M. J. Laudeman
S. T. Shipley

CLERICAL STAFF

T. T. Claiborne
D. A. Cox
R. D. Ellis
M. S. Perry
K. E. Waters

ILLUSTRATOR

E. A. Powell

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS AND ACRONYMS

ASME	American Society of Mechanical Engineers
cm^2	square centimeter
cpm	counts per minute
dpm/100 cm^2	disintegrations per minute/100 square centimeters
EPA	Environmental Protection Agency
EML	Environmental Measurement Laboratory
ESSAP	Environmental Survey and Site Assessment Program
ft	foot
GM	Geiger-Mueller
kg	kilogram
kW	kilowatt
m	meter
m^2	square meter
MDA	Minimum Detectable Activity
NaI	Sodium Iodide
NES	Nuclear Energy Services
NIST	National Institute for Standards Technology
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
PIC	Pressurized Ionization Chamber
pCi/g	picocuries per gram
$\mu R/h$	microroentgen per hour
UCLA	University of California at Los Angeles
ZnS	zinc sulfide

CONFIRMATORY SURVEY
OF THE
BOELTER REACTOR FACILITY
UNIVERSITY OF CALIFORNIA AT
LOS ANGELES, CALIFORNIA

INTRODUCTION AND SITE HISTORY

The Boelter Reactor Facility is owned by the University of California at Los Angeles (UCLA). The facility contained a 100 kW Argonaut type research reactor which was operated between October 1960 and June 1984 under U.S. Nuclear Regulatory Commission (NRC) License R-71 [Docket 50-142], for research and training purposes. The reactor fuel utilized during operation was 93% enriched uranium. There are no records of any major incidents involving releases of radioactive material during the operating history of the reactor. The university has completed the dismantlement of the reactor and remediation of the facility and plans to release the site for unrestricted use.

Nuclear Energy Services (NES), contracted by UCLA, carried out the decommissioning activities in two phases. The decommissioning plan for Phase I, approved by NRC in July 1986, included the removal of the reactor core-moderator, graphite thermal column, shield tank, peripheral equipment, fuel boxes, control blade system components, protruding parts of pipes and structures, and most of the concrete shield blocks. The decommissioning plan for Phase II, approved by NRC in July 1989, included the demolition of the reactor monolith, removal of the remaining shield blocks, removal of all process equipment, and decontamination of all remaining facilities. The licensee's laboratory analysis of the concrete from the interior monolith wall and one of the removable blocks had indicated that the primary contaminants were Co-60 and Eu-152—products of neutron activation due to reactor operations.

A final radiological survey of the facility was performed by the licensee and the results were provided to the NRC in December 1992.[1] The U.S. Nuclear Regulatory Commission, Region V Office, has requested that the Environmental Survey and Site Assessment Program (ESSAP) of Oak Ridge Institute for Science and Education (ORISE) perform an independent confirmatory survey of this facility.

SITE DESCRIPTION

The facility is located on the UCLA campus in Los Angeles, California on the ground floor of Boelter Hall (Figures 1 and 2). The building is a two story structure. The exterior walls of the reactor facility are load bearing and are of 12-inch thick, reinforced concrete. The interior walls separating the reactor high bay from remainder of the building are also load bearing and are of reinforced 18-inch thick concrete. All other interior walls are nonbearing, 8-inch, cement block walls.

In the reactor high bay area, a significant portion of the floor area has been pitted due to removal of the reactor monolith. A subfloor excavation, approximately 1.5 meters deep, is located in the northeast corner of the pitted area. North of where the reactor was located, there is a process pit with an extension under the floor to the west which contained two retention tanks. There are also 30 galvanized steel-lined fuel storage pits buried in the concrete (Figure 3).

The storage room (1003), counting room (1005) and a sloped ramp area with a combined floor space of approximately 100 m^2 are located on the first floor adjacent to the reactor high bay area (Figure 3). On the second floor, the control room (2001) and the adjacent locker room (or the change room) are also included in the decommissioning activities. The control room is less than 50 m^2 and contains a tiled floor. The change room has primarily ceramic surfaces. The second floor also includes a balcony walkway and a section of expanded metal catwalk encircling the reactor high bay area (Figure 4).

No other rooms in the facility and no areas outside Boelter Hall were within the scope of the decommissioning activities.

OBJECTIVES

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

DOCUMENT REVIEW

As part of the confirmatory activities, ESSAP reviewed the licensee's radiological survey data.[1] Analytical procedures and methods utilized by the licensee were reviewed for adequacy and appropriateness. The data were reviewed for accuracy, completeness, and compliance with guidelines.

PROCEDURES

On March 10 and 11, 1993, ORISE performed a confirmatory radiological survey of the Boelter Reactor Facility at the University of California at Los Angeles. The survey was conducted in accordance with a survey plan which was submitted to and approved by the NRC Region V Office.[2] This report summarizes the procedures and results of the survey.

REFERENCE GRID

The reference grid system established by the licensee was used by ESSAP to reference measurement and sampling locations (Figures 5 through 11). Depending on the area, the grid size was 1 m × 1 m, 2 m × 2 m, or 3 m × 3 m.

SURFACE SCANS

Surface scans for alpha, beta, and gamma activity were performed on floors and lower walls (up to 2 m), using large area gas proportional and NaI detectors coupled to ratemeter-scalers and

ratemeters with audible indicators. Three drains in the reactor room, the accessible surfaces of the ventilation duct, and the steel-lined fuel storage pits were also scanned.

SURFACE ACTIVITY MEASUREMENTS

Direct measurements to determine total alpha and total beta surface activity were performed on 56 randomly selected grid blocks on the floor and lower walls in each room and on the balcony. Measurements were performed at the center and at four points equidistant from the center and grid block corners. Eight single-point measurements were performed on upper wall surfaces. A smear sample for determining removable activity was obtained from each grid block at the location corresponding to the maximum total activity and from each single-point measurement location. Smear samples for determining H-3 and C-14 activity were collected from the center point of 18 of the randomly selected grid blocks. Measurement and sampling locations for total and removable activity are illustrated in Figures 5 through 11.

EXPOSURE RATE MEASUREMENT

Background exposure rate measurements were performed at 3 building locations having similar construction as the Boelter facility but without a history of radioactive materials use. Exposure rate measurements were performed at 1 m (3.3 ft) above the interior building surfaces at 8 locations using a pressurized ionization chamber (PIC). Measurement locations are shown in Figures 5 through 11.

MISCELLANEOUS SAMPLING

Two samples were collected in the reactor monolith pitted area; a soil sample was collected from the subfloor excavated area, and a concrete sample was collected from hole #2, one of the three existing holes in the concrete pedestal. Another concrete sample was collected from the same general area, at a location where elevated gamma activity was detected by surface scans (Figure 5). In addition, at the request of the NRC, the licensee provided ESSAP with five samples for

confirmatory analysis; three soil samples from underneath the concrete pedestal, and two samples from the concrete cap in hole #2.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and survey data were returned to the ESSAP Oak Ridge laboratory for analyses and interpretation. Smears were analyzed for gross alpha and gross beta or H-3/C-14 activity. Direct measurement and smear data were converted to units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2), and exposure rate measurements were reported in microroentegens per hour (μR/h). Soil and concrete samples were analyzed by gamma spectrometry. Spectra were reviewed for U-235, U-238, Co-60, and Eu-152 and any other identifiable photopeaks. Soil and concrete samples results were reported in units of picocuries per gram (pCi/g). Additional information concerning major instrumentation, sampling equipment, and analytical procedures is provided in Appendices A and B. Results were compared to NRC guidelines which are provided in Appendix C.

FINDINGS AND RESULTS

DOCUMENT REVIEW

ESSAP reviewed the licensee's radiological survey data and comments were provided to the NRC.[3] In ESSAP's opinion, the document provides an adequate description of the radiological condition of the facility relative to the NRC guidelines for release to unrestricted use.

SURFACE SCANS

Surface scans for alpha, beta, and gamma activity of the floor and lower walls identified one location of elevated direct gamma radiation slightly in excess of ambient background levels in the southwest corner of the reactor monolith pit. A concrete sample was taken from this location for laboratory analysis.

Three drains, two in the processing pit and one in the southeast quadrant of the reactor room were scanned. No locations of elevated direct radiation were noted. The fuel storage pits and the accessible surfaces of the ventilation duct in the reactor room and the outlet on the roof were also scanned. No locations of elevated direct radiation were noted.

SURFACE ACTIVITY LEVELS

Results of total and removable activity are summarized in Table 1. Total activity ranged from <140 to 160 dpm/100 cm^2 for alpha and from <2100 to 2900 dpm/100 cm^2 for beta. Grid block averages were all less than the detection limits of the procedure which were 140 dpm/100 cm^2 and 2100 dpm/100 cm^2 for alpha and beta, respectively. Removable activity levels were less than the minimum detectable activity of the procedure which were 12 dpm/100 cm^2 for alpha, 17 dpm/100 cm^2 for beta, and 6 dpm/100 cm^2 for both H-3 and C-14.

EXPOSURE RATES

Exposure rate measurements from 8 locations within the Boelter Reactor Facility and 3 other locations are summarized in Table 2. Background exposure rates ranged from 12 to 13 μR/h and averaged 12 μR/h. Boelter Reactor Facility exposure rates ranged from 13 to 14 μR/h.

MISCELLANEOUS SAMPLES

Radionuclide concentrations for the soil and concrete samples collected from the reactor concrete pedestal are presented in Table 3. For the four soil samples, collected from underneath the concrete pedestal and in the subfloor excavation north of the concrete pedestal, the concentration of Co-60 was $\leq$1.0 pCi/g. The concentration of Eu-152 ranged from <0.1 to 3.8 pCi/g. The concentration of U-235 was 0.1 pCi/g and the concentration of U-238 ranged from 0.4 to 1.4 pCi/g.

For the three concrete samples from hole #2 of the concrete pedestal, the concentrations of Co-60 and Eu-152 ranged from 0.1 to 3.3 pCi/g and 0.6 to 11.3 pCi/g, respectively. The concentrations of U-235 and U-238 were 0.1 pCi/g and $\leq$1.0 pCi/g, respectively.

For the concrete sample, collected at the edge of the monolith pit where slightly elevated gamma activity was noted, the concentrations of Co-60 and Eu-152 were 1.4 and 7.8 pCi/g, respectively. The concentrations of U-235 and U-238 were 0.1 and 1.0 pCi/g, respectively.

COMPARISON OF RESULTS WITH GUIDELINES

The NRC guidelines for surface contamination and residual concentrations of radionuclides in soil, established for license termination or release of a facility for unrestricted use, are presented in Appendix C.[4,6] The major contaminants identified were Co-60 and Eu-152. The applicable guidelines are those for beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 are:

Total Activity

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

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

Removable Activity

1000 dpm β-γ/100 cm^2

The surface contamination guidelines for uranium are:

Total Activity

5,000 dpm α/100 cm^2, averaged over a 1 m^2 area

15,000 dpm α/100 cm^2, maximum in a 100 cm^2 area

Removable Activity

1,000 dpm α/100 cm^2

All surface activity measurements were within the guideline levels.

Exposure rates throughout the facility were all within the range of background radiation and, therefore, below the guideline level of 5 μR/h above background.[5]

The soil concentration guideline for enriched uranium is 30 pCi/g.[6] Based on a U-234:U-235 activity ratio of 40 to 1, the highest total uranium concentration in the samples collected was 5.5 pCi/g which is well below the 30 pCi/g limit.

There are no specific concentration guidelines for Co-60 and Eu-152 in soil. However, in similar facilities an exposure rate limit of 5 μR/h above background, measured at one meter above surface, has been used to limit the concentrations of Co-60 and Eu-152 in concrete, components, structures, and soil.[7] In this survey, the highest concentrations of Co-60 (3.6 pCi/g) and Eu-152 (16.4 pCi/g) were identified at hole #2 of the concrete pedestal. ESSAP's review of the licensee's final decommissioning report indicates that the exposure rates, at one meter above surface, at the three holes in the concrete pedestal and at the southwest corner of the monolith pit are less than 5 μR/h above background. This is consistent with ESSAP exposure rate measurements at other locations in the reactor room.

SUMMARY

On March 10 and 11, 1993, ESSAP performed a confirmatory survey of the Boelter Reactor Facility, on the campus of the University of California at Los Angeles, California. Survey activities included document reviews, surface scans, measurements of total and removable surface activity, exposure rate measurements and soil and concrete sampling.

In ESSAP's opinion, the licensee's documentation provides an adequate description of the radiological condition of the Boelter Reactor Facility. ESSAP confirmatory measurements support the licensee's conclusion that the facility satisfies NRC's guidelines for release to unrestricted use.

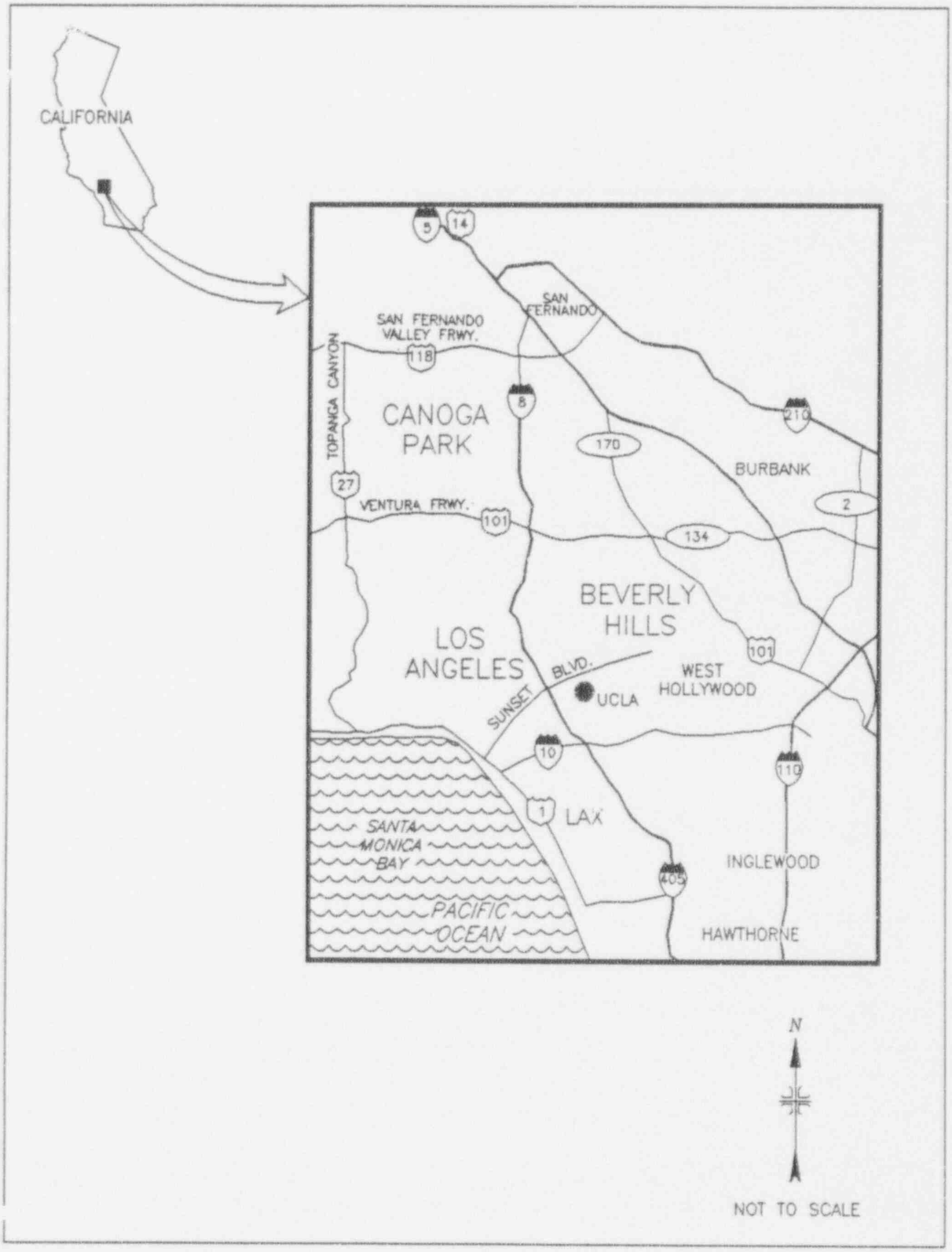


FIGURE 1: Los Angeles, California Area, Location of the University of California at Los Angeles Campus

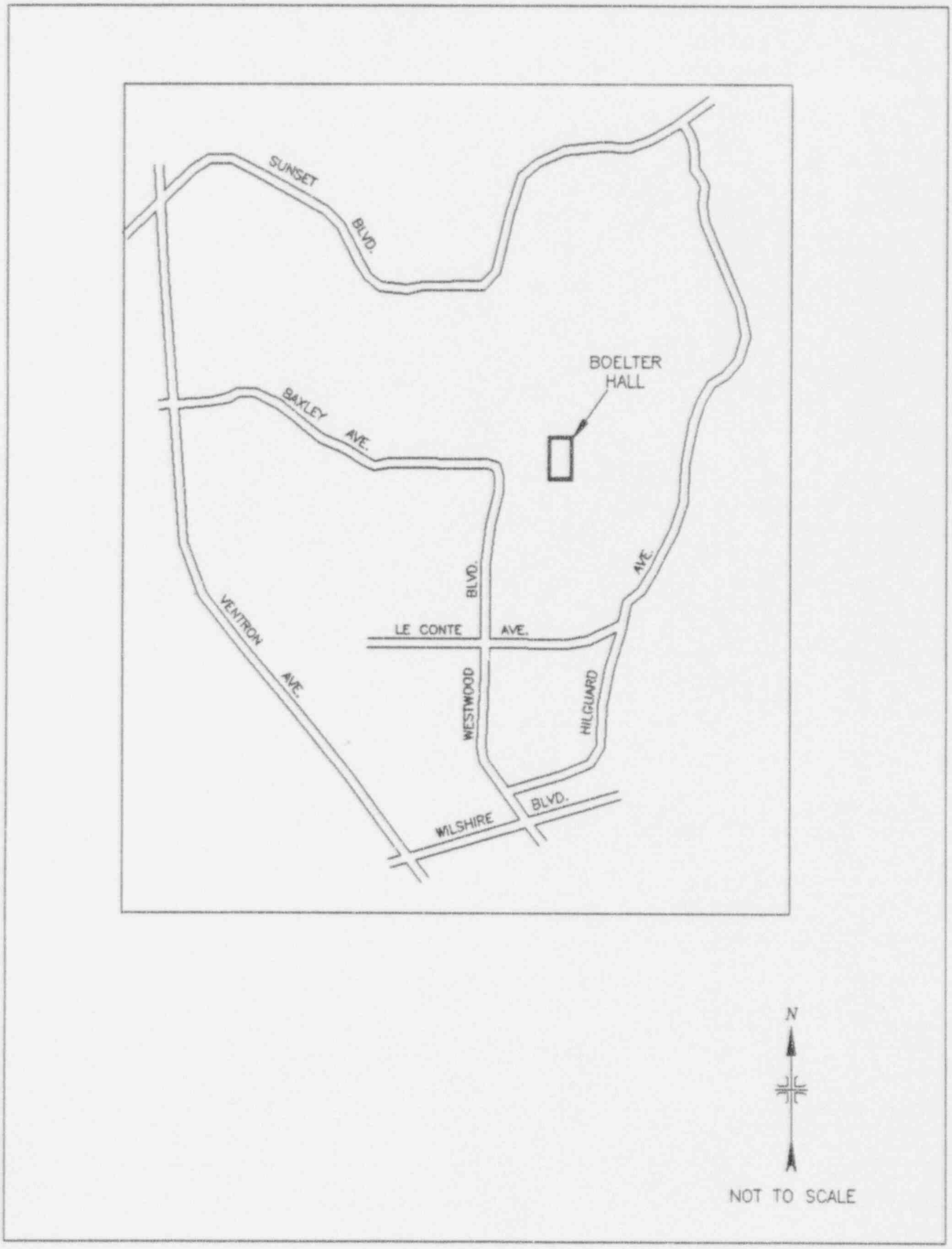


FIGURE 2: University of California at Los Angeles Campus – Location of Boelter Hall

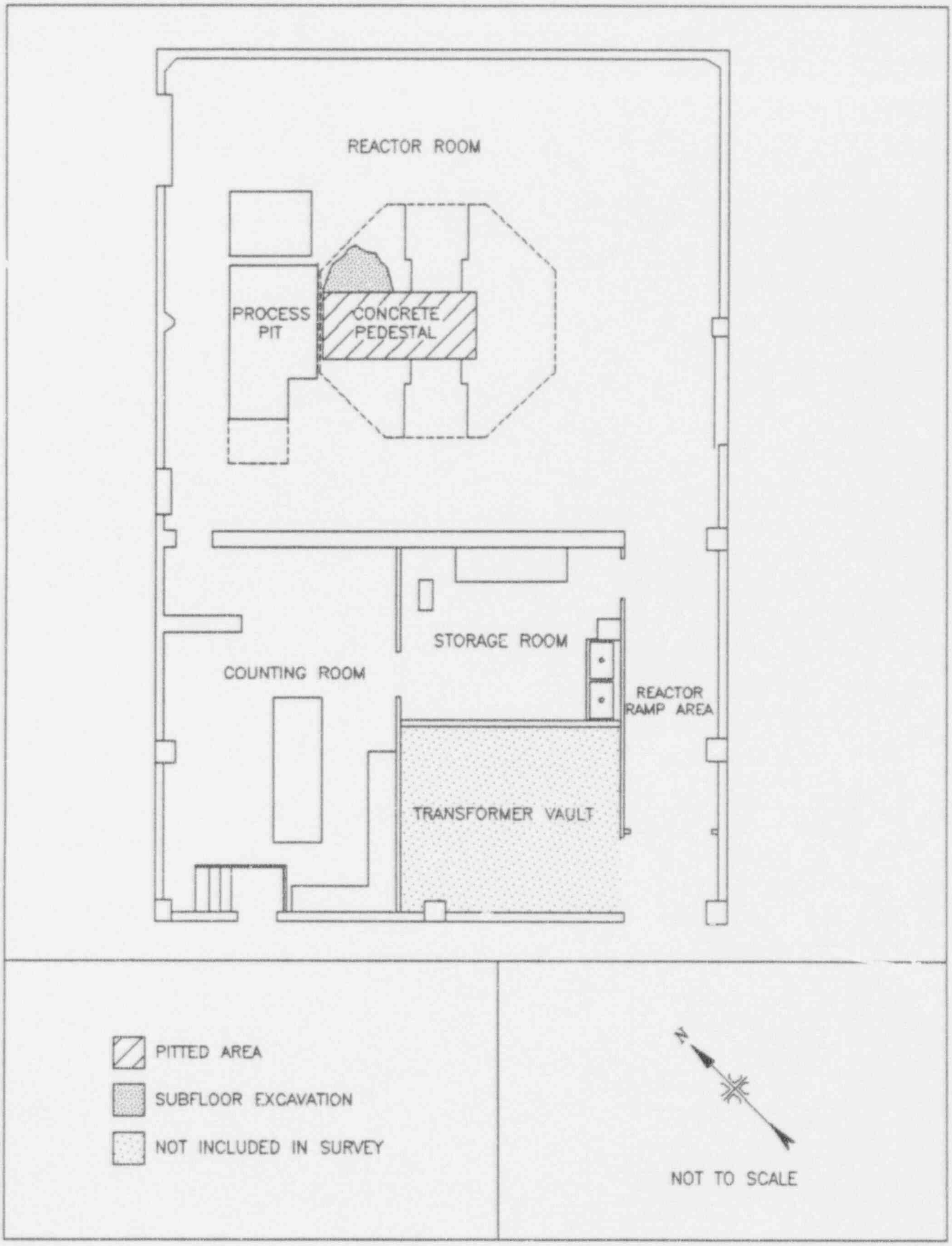


FIGURE 3: Boelter Reactor Facility, Plot Plan of Ground Floor

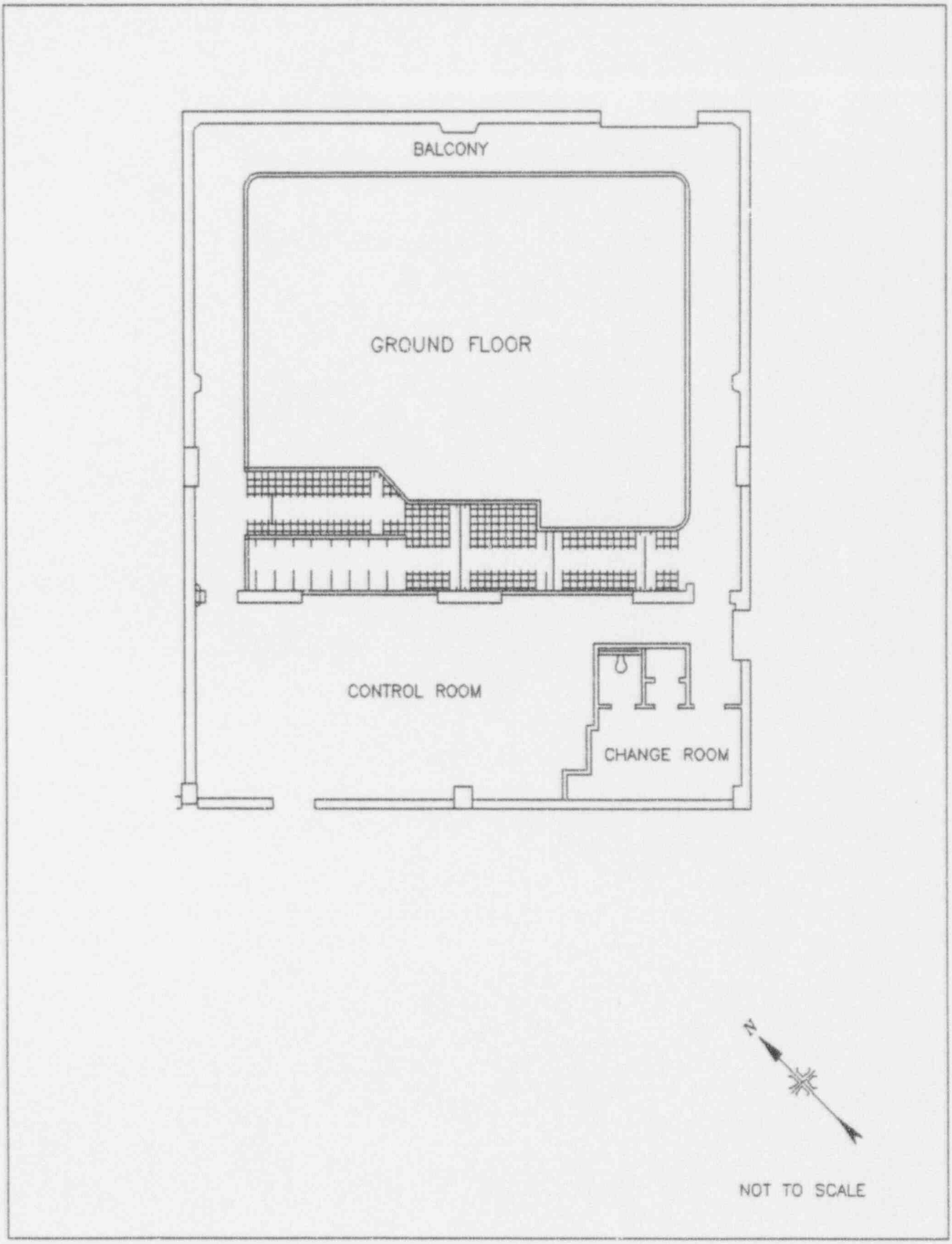


FIGURE 4: Boelter Reactor Facility, Plot Plan of Second Floor

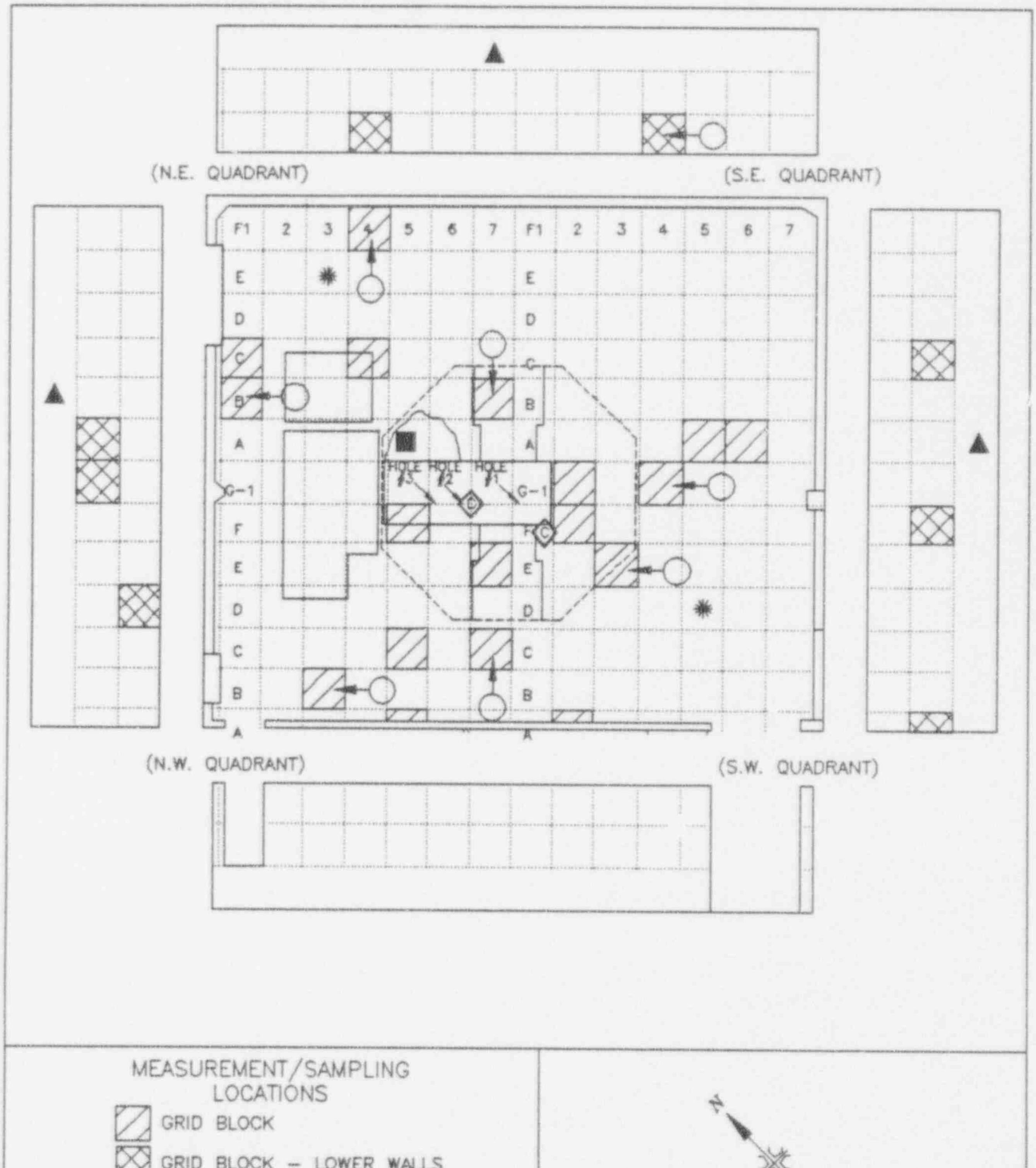


FIGURE 5: Reactor Room - Measurement and Sampling Locations

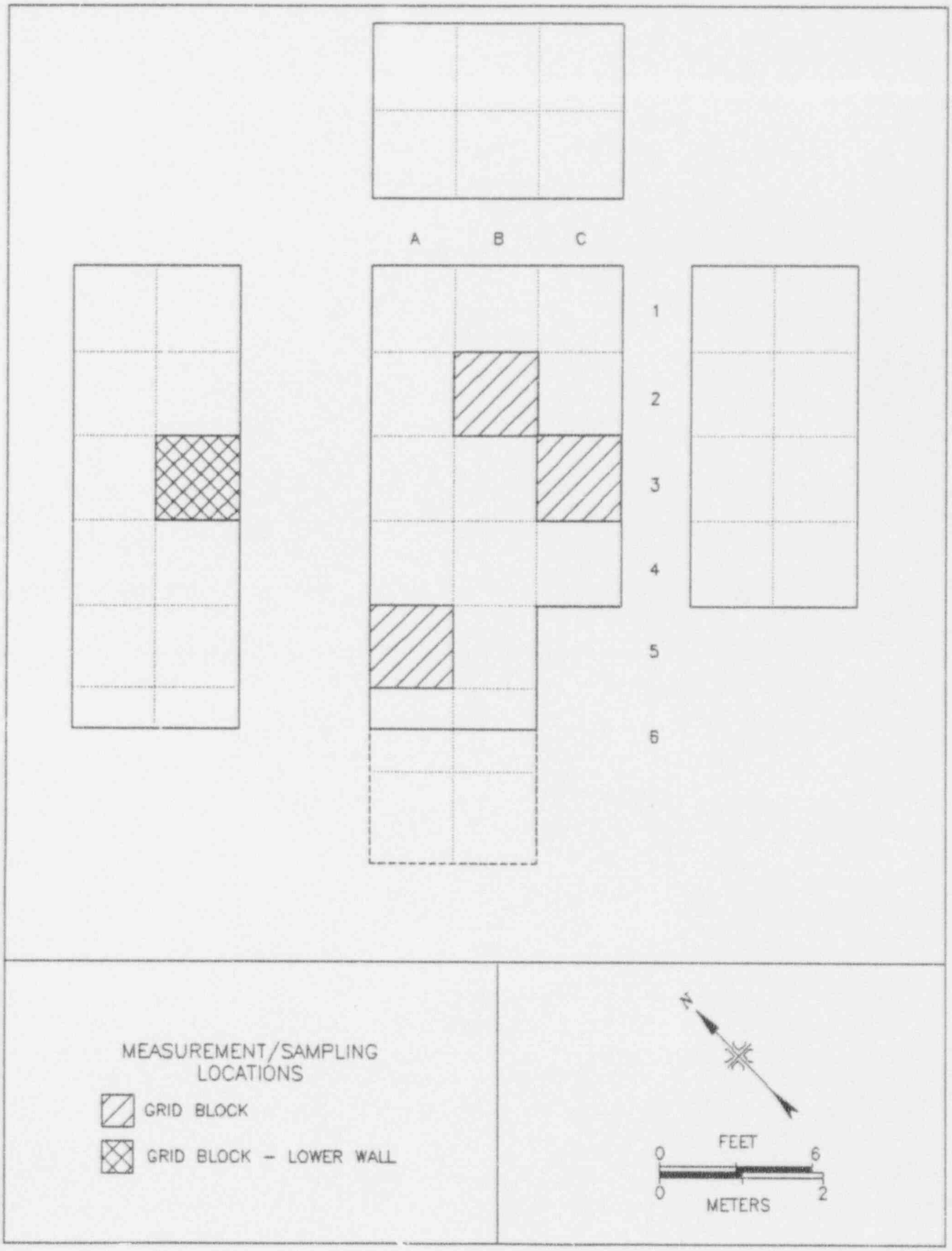


FIGURE 6: Reactor Room Process Pit – Measurement and Sampling Locations

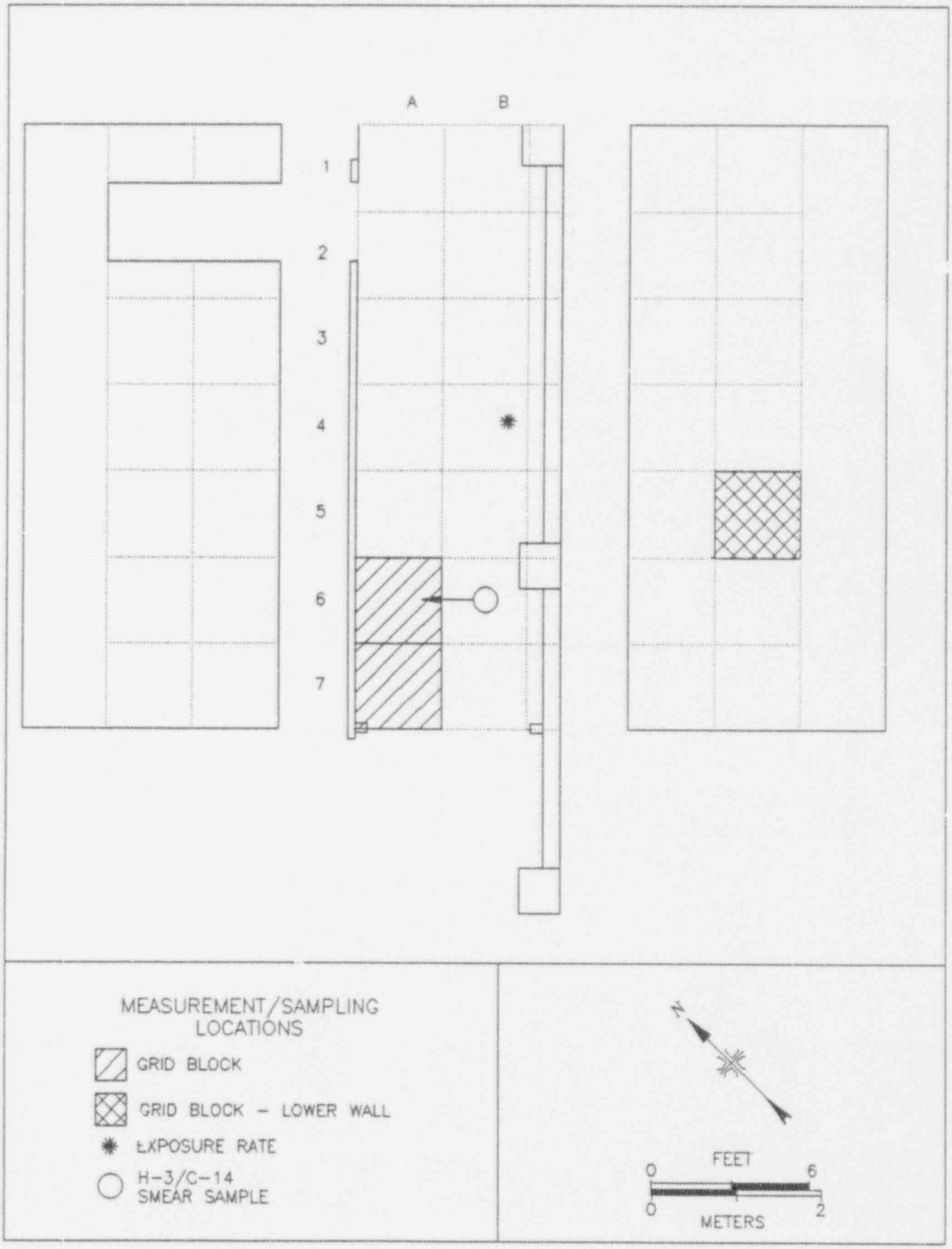


FIGURE 7: Reactor Ramp Area – Measurement and Sampling Locations

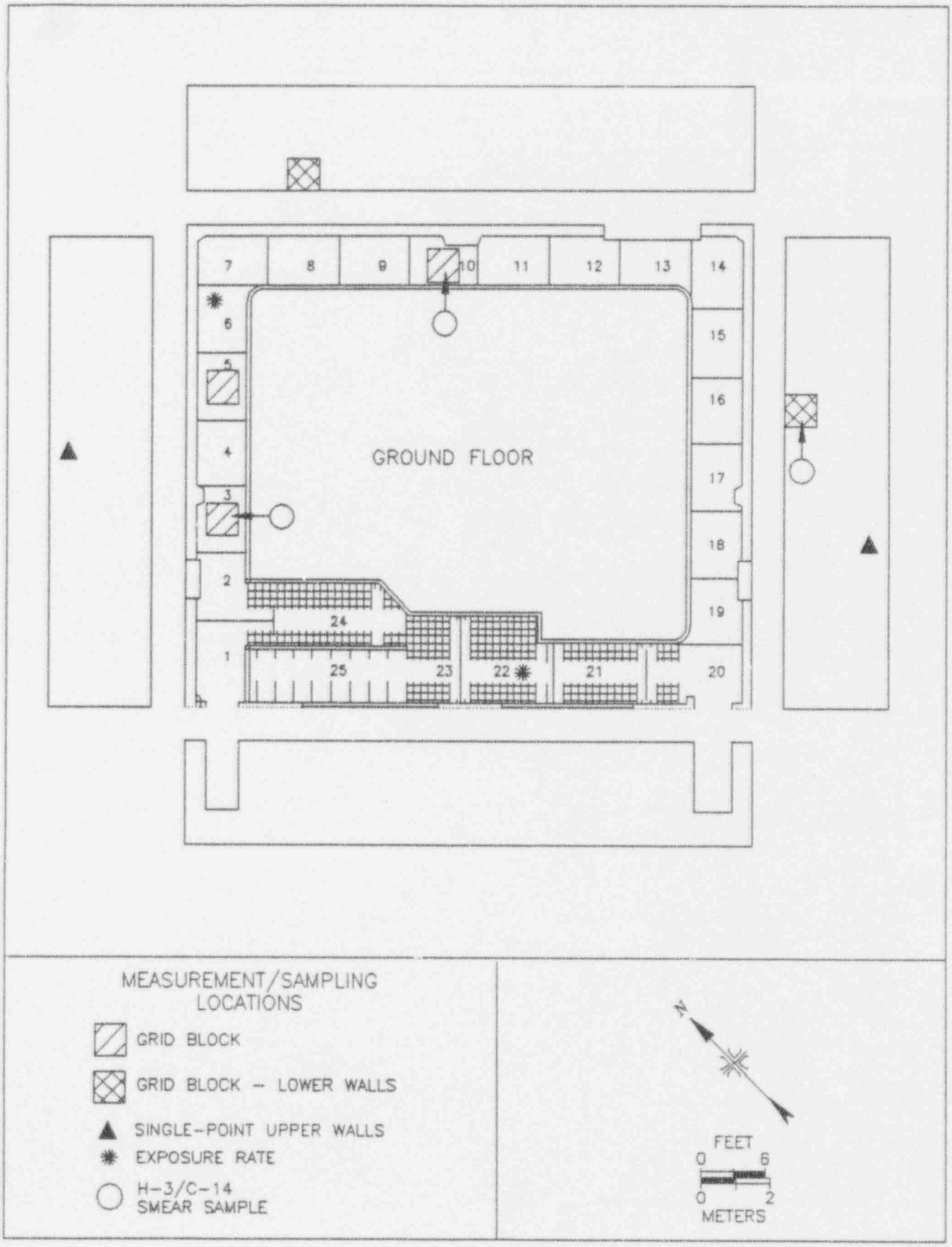


FIGURE 8: Reactor Room Balcony – Measurement and Sampling Locations

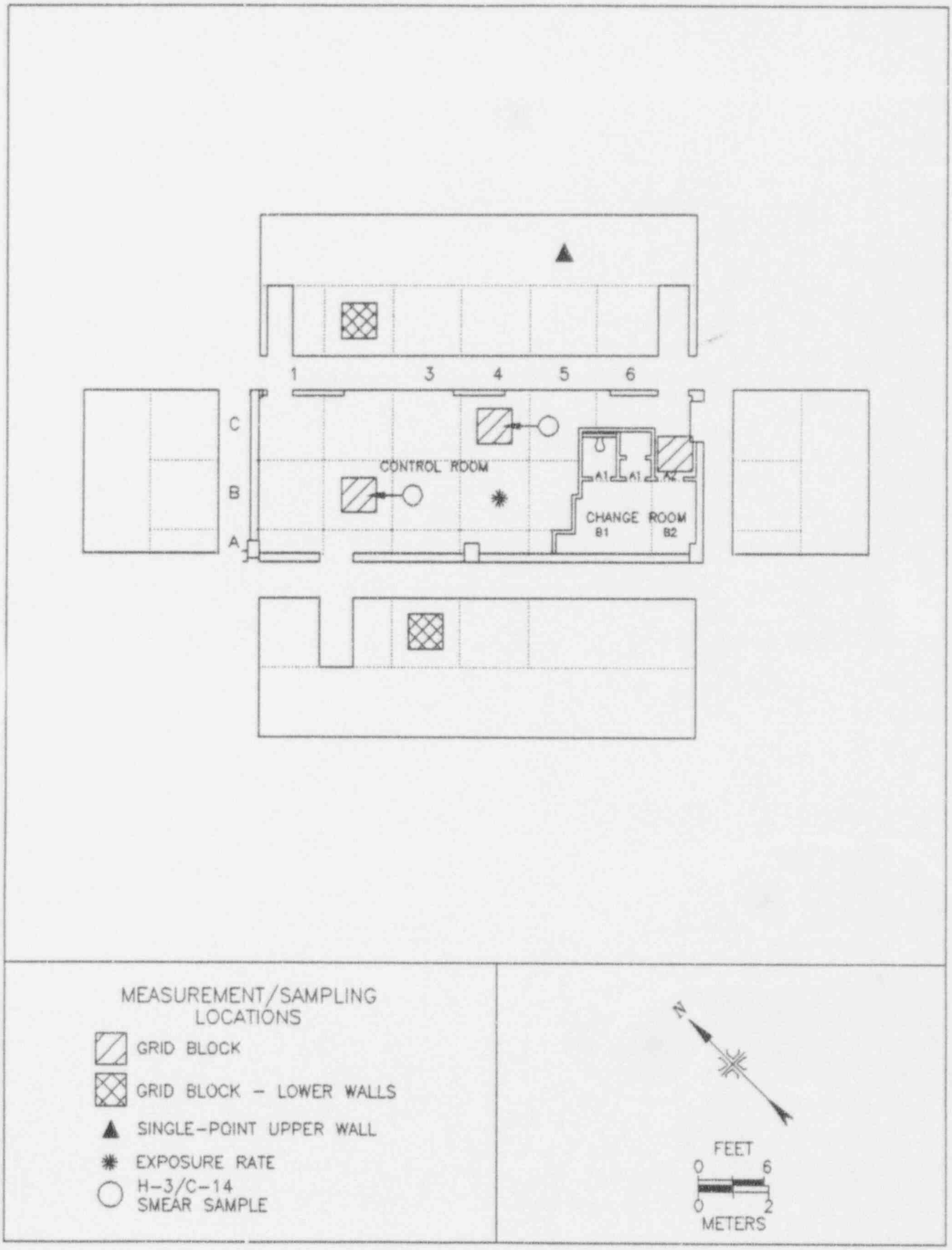


FIGURE 9: Control Room and Change Room - Measurement and Sampling Locations

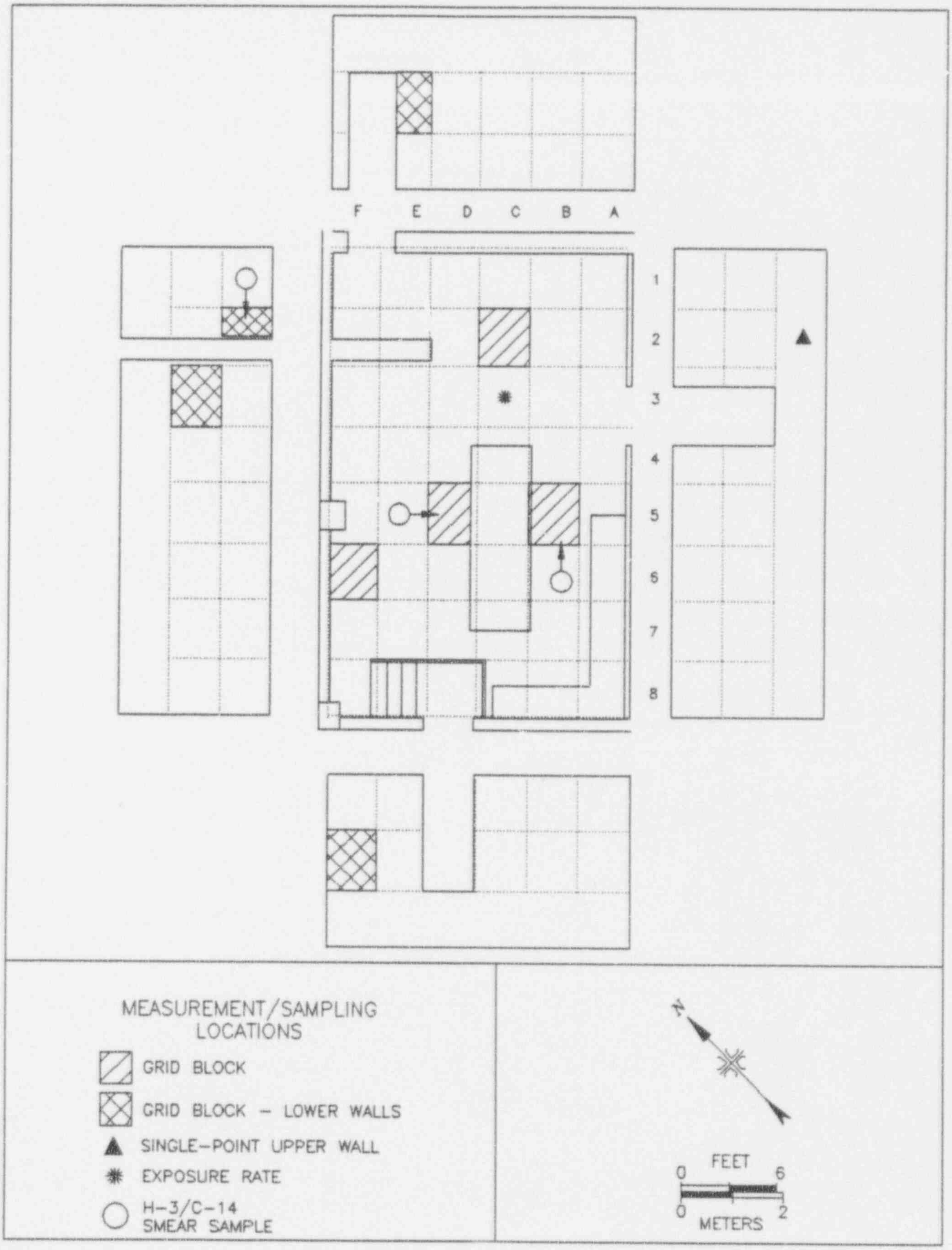


FIGURE 10: Counting Room - Measurement and Sampling Locations

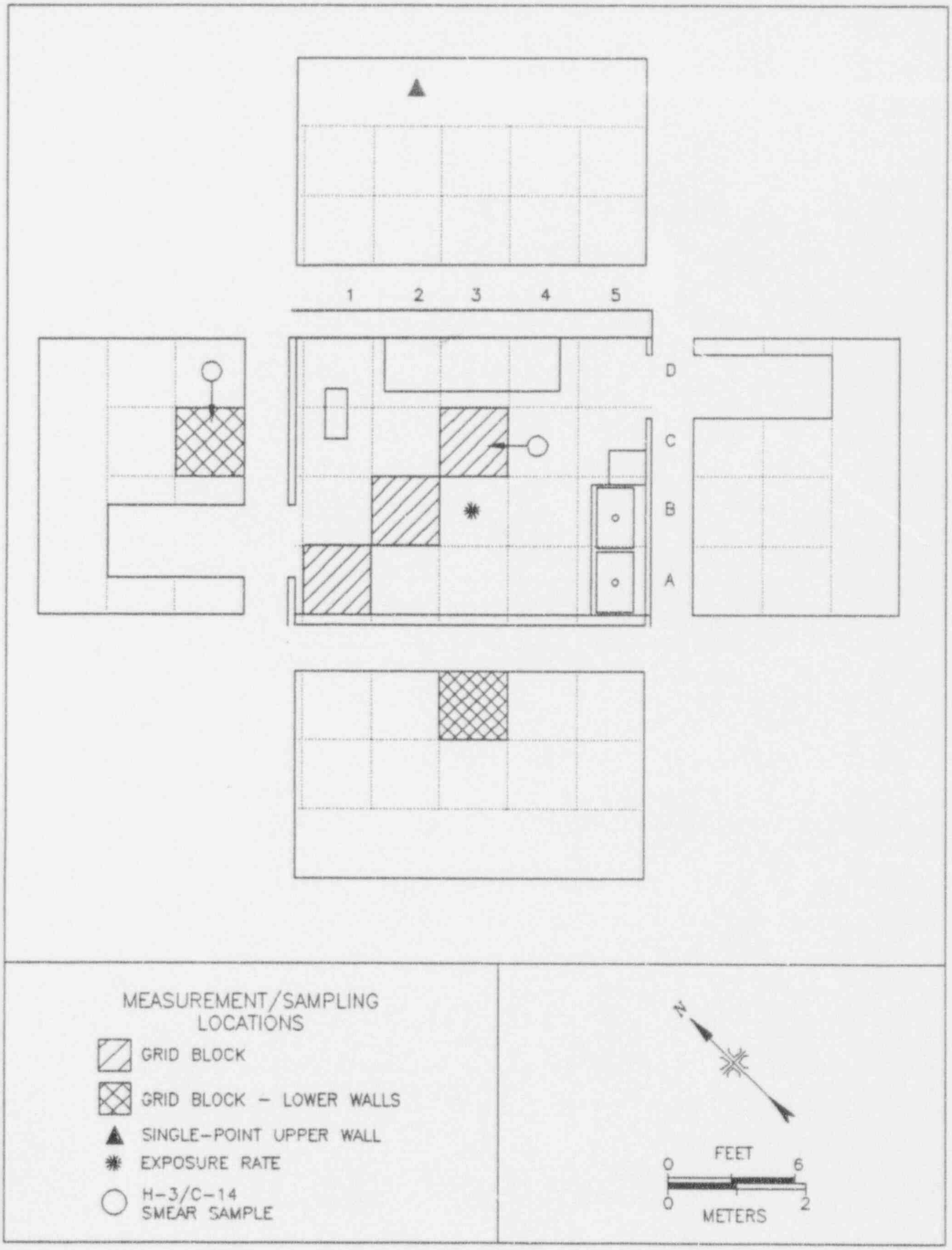


FIGURE 11: Storage Room - Measurement and Sampling Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY MEASUREMENTS THE BOELTER REACTOR FACILITY UNIVERSITY OF CALIFORNIA AT LOS ANGELES, CALIFORNIA

Location	Figure Number	Number of Grid Block/ Single Pt. Measurements	Range of Total Activity (dpm/100 cm^2)				Range of Removable Activity (dpm/100 cm^2)	
			Single Measurement		Grid Block Average			
			Alpha	Beta	Alpha	Beta	Alpha	Beta
Reactor Room								
Lower Walls and Floor	5	26/0	<140-160	<2100-2900	<140	<2100	<12	<17
Upper Walls		0/3	<140	<2100	N/A[a]	N/A	<12	<17
Reactor Room Process Pit								
Lower Walls and Floor	6	4/0	<140	<2100	<140	<2100	<12	<17
Reactor Room Ramp								
Lower Walls and Floor	7	3/0	<140	<2100	<140	<2100	<12	<17
Reactor Room Balcony								
Lower Walls and Floor	8	5/0	<140	<2100	<140	<2100	<12	<17
Upper Walls		0/2	<140	<2100	N/A	N/A	<12	<17

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY MEASUREMENTS
THE BOELTER REACTOR FACILITY
UNIVERSITY OF CALIFORNIA AT
LOS ANGELES, CALIFORNIA

Location	Figure Number	Number of Grid Block/ Single Pt. Measurements	Range of Total Activity (dpm/100 cm²)				Range of Removable Activity (dpm/100 cm²)	
			Single Measurement		Grid Block Average			
			Alpha	Beta	Alpha	Beta	Alpha	Beta
Control Room								
Lower Walls and Floor	9	5/0	<140	<2100-2500	<140	<2100	<12	<17
Upper Walls		0/1	<140	<2100	N/A	N/A	<12	<17
Counting Room								
Lower Walls and Floor	10	8/0	<140	<2100-2700	<140	<2100	<12	<17
Upper Walls		0/1	<140	<2100	N/A	N/A	<12	<17
Storage Room								
Lower Walls and Floor	11	5/0	<140	<2100	<140	<2100	<12	<17
Upper Walls		0/1	<140	<2100	N/A	N/A	<12	<17

[a]Not applicable, grid block measurements were not performed on upper (>2 m) surfaces.

TABLE 2

EXPOSURE RATE MEASUREMENTS
THE BOELTER REACTOR FACILITY
UNIVERSITY OF CALIFORNIA AT
LOS ANGELES, CALIFORNIA

Location	Figure	Exposure Rate at 1 m above surface (μR/h)
Background Locations		
Boelter Hall Entrance Hallway	N/A	13
Boelter Hall, 2nd Floor Hallway	N/A	12
Building Adjacent to Boelter Hall	N/A	12
Reactor Facility Locations		
Reactor Room Grid D5, SW	5	13
Reactor Room Grid E3, NE	5	13
Reactor Room Ramp Grid B4	7	13
Balcony Grid #6	8	13
Balcony Grid #22	8	13
Control Room Grid B4	9	13
Counting Room Grid C3	10	13
Storage Room Grid B3	11	14

TABLE 3

RADIONUCLIDE CONCENTRATIONS IN MISCELLANEOUS SAMPLES THE BOELTER REACTOR FACILITY UNIVERSITY OF CALIFORNIA AT LOS ANGELES, CALIFORNIA

Sample Type	Sample Location	Radionuclide Concentrations (pCi/g)[a]			
		Co-60	Eu-152	U-235	U-238
Concrete	hole #2 concrete pedestal	3.3 ± 0.1	11.3 ± 0.2	0.1 ± 0.1	0.1 ± 0.4
Soil	subfloor excavation grid A5, NE	0.8 ± 0.1	3.8 ± 0.1	0.1 ± 0.1	0.4 ± 0.1
Concrete	monolith pit grid F1, SW	1.4 ± 0.3	7.8 ± 0.5	0.1 ± 0.1	1.0 ± 1.0
Soil[b]	hole #1 concrete pedestal	<0.1	0.1 ± 0.1	0.1 ± 0.1	1.0 ± 0.6
Soil[b]	hole #2 concrete pedestal	0.1 ± 0.1	<0.1	0.1 ± 0.1	1.4 ± 0.6
Soil[b]	hole #3 concrete pedestal	0.1 ± 0.1	<0.1	0.1 ± 0.1	0.9 ± 0.7
Powdered[b] concrete	hole #2 concrete pedestal	2.0 ± 0.1	7.5 ± 0.1	0.1 ± 0.1	0.4 ± 0.4
Chipped[b] concrete	hole #2 concrete pedestal	0.1 ± 0.1	0.6 ± 0.2	0.1 ± 0.1	1.0 ± 1.0

[a]Uncertainties represent the 95% confidence level, based only on counting statistics.
[b]These sample were provided to ESSAP by the licensee for confirmatory analysis.

REFERENCES

1. University of California, Los Angeles, California, "Final Decommissioning Report for the Boelter Hall Reactor Facility Dismantlement and Final Decommissioning Project," December 1992.

2. Oak Ridge Institute for Science and Education, "Confirmatory Radiological Survey Plan for the Boelter Reactor Facility, University of California, Los Angeles, California," February 22, 1993.

3. Letter from M. R. Landis (ORISE) to M. Cillis (NRC, Region V), Reference: "Final Decommissioning Report for Boelter Reactor Facility Dismantlement and Final Decommissioning Project, University of California," January 28, 1993.

4. U.S. Nuclear Regulatory Commission, "Termination of Operating Licenses for Nuclear Reactors," Regulatory Guide 1.86, Washington D.C., June 1974.

5. U.S. Nuclear Regulatory Commission, Office of Nuclear Safety and Safeguards, "Review Plan: Evaluating Decommissioning Plans for Licenses Under 10 CFR Parts 30, 40, and 70," Washington, D.C. 1991.

6. U.S. Nuclear Regulatory Commission, "Disposal or Onsite Storage of Thorium and Uranium Wastes from Past Operations", 46 FR 52061, Washington, D.C., October 23, 1981.

7. Letter from J. Stolz (NRC) to R. Finston (Stanford University), Reference: Docket No. 50-141, enclosure 1, March 17, 1981.

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

<u>Instruments</u>

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)

<u>Detectors</u>

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)

Ludlum Gas Proportional Detector
Model 43-68
Effective Area, 100 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-5100-W
(Oxford, Oak Ridge, TN)

Tri-Carb Liquid Scintillation Analyzer
Model 1900CA
(Packard Instrument Co., Meriden, CT)

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 of the surveyed areas. Other surfaces were scanned using small area (100 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-Beta	-	gas proportional detector with ratemeter-scaler
Gamma	-	NaI scintillation detector with ratemeter

Surface Activity Measurements

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

Count rates (cpm), which were integrated over 0.5 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 averaged approximately 1 cpm for each detector. Alpha efficiency factors ranged from 0.18 - 0.19 for the ZnS scintillation detectors. The beta activity background count rates for the GM detectors averaged 48 cpm. Beta efficiency factors averaged 0.16 for the GM

detector. The effective windows 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.

Two smear samples for removable contamination (one smear for gross alpha/beta and one smear for H-3/C-14 analyses) were obtained from each measurement location.

Exposure Rate Measurements

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

Miscellaneous Samples

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.

Concrete Sampling

The concrete sample was taken by chipping material from approximately 100 cm^2 of surface.

ANALYTICAL PROCEDURES

Removable Activity

Gross Alpha/Beta

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

Liquid Scintillation

Smears were counted in a liquid scintillation counter for low-energy beta activity to determine H-3 and C-14 activity.

Miscellaneous Samples

Gamma Spectrometry

Solid Samples

Samples of solid material (soil and construction material) were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in 0.5-liter Marinelli beaker or other appropriate container. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry. 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
Eu-152	0.344 MeV

U-235	0.186 MeV
U-238	0.063 MeV from Th-234*

*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. Additional uncertainties, associated with sampling and measurement procedures, have not been propagated into the data presented in this report.

Detection limits, referred to as minimum detectable activity (MDA), were based on 2.71 plus 4.66 times the standard deviation of the background count. When the activity was determined to be less than the MDA of the measurement procedure, the result was reported as less than MDA. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclide in samples, the detection limits differ from sample to sample and instrument to instrument.

CALIBRATION AND QUALITY ASSURANCE

Analytical and field survey activities were conducted in accordance with procedures from the following documents:

- 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, when such standards/sources were available. In cases where they were not available, standards of an industry recognized organization was used. Calibration of pressurized ionization chambers was performed by the manufacturer.

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 laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

APPENDIX C

REGULATORY GUIDE 1.86, TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

AND

GUIDELINES FOR RESIDUAL CONCENTRATIONS OF THORIUM AND URANIUM WASTES IN SOIL

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 from the reactor and placing it in storage (either onsite or 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 [illegible]ted.

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 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 off-site for burial at an authorized burial ground or

USAEC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the AEC regulatory staff of implementing specific parts of the Commission's regulations, to originate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Published guides will be revised periodically, as appropriate, to accommodate comments and to reflect new information or experience.

Copies of published guides may be obtained by request indicating the division desired to the U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Director of Regulatory Standards. Comments and suggestions for improvements in these guides are encouraged and should be sent to the Secretary of the Commission, U.S. Atomic Energy commission, Wash[illegible]ngton, D.C. 20545, Attention: Chief, Public Proceedings Staff.

The guides are issued in the following ten broad divisions.

1. Power Reactors
2. Research and test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plan Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust Review
10. General

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 1).

The hazard associated with the returned 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 1 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 1) 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 must 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 of 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 1. 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 2contamination 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 permissible 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 1. 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 1.

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 1, the Commission may terminate the license. If 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 1

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	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 pert ient levels should be reduced proportionally and the entire surface should be wiped.

Note: Section electronically reproduced from photocopy.

Guidelines for Residual Concentrations of Thorium and Uranium Wastes in Soil

On October 23, 1981, the Nuclear Regulatory Commission published in the Federal register a notice of Branch Technical Position on "Disposal or Onsite Storage of Thorium and Uranium Wastes from Past Operations." This document established guidelines for concentrations of uranium and thorium in soil, that will limit maximum radiation received by the public under various conditions of future land usage. These concentrations are as follows:

Material	Maximum Concentrations (pCi/g) for various options			
	1[a]	2[b]	3[c]	4[d]
Natural Thorium (Th-232 + Th-228) with daughters present and in equilibrium	10	50	--	500
Natural Uranium (U-238 + U-234) with daughters present and in equilibrium	10	--	40	200
Depleted Uranium:				
Soluble	35	100	--	1,000
Insoluble	35	300	--	3,000
Enriched Uranium:				
Soluble	30	100	--	1,000
Insoluble	30	250	--	2,500

[a]Based on EPA cleanup standards which limit radiation to 1 mrad/yr to lung and 3 mrad/yr to bone from ingestion and inhalation and 10 μR/h above background from direct external exposure.
[b]Based on limiting individual dose to 170 mrem/yr.
[c]Based on limiting equivalent exposure to 0.02 working level or less.
[d]Based on limiting individual dose to 500 mrem/yr and in case of natural uranium, limiting exposure to 0.02 working level or less.