CONFIRMATORY SURVEY OF THE TURBINE BUILDING, SITE GROUNDS, AND SITE EXTERIORS SHOREHAM NUCLEAR POWER STATION BROOKHAVEN, NEW YORK

[DOCKET No. 50-322]

T. J. VITKUS

Prepared for the
Division of Waste Management
Headquarters Office
U.S. Nuclear Regulatory Commission

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OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

Environmental Survey and Site Assessment Program
Energy/Environment Systems Division

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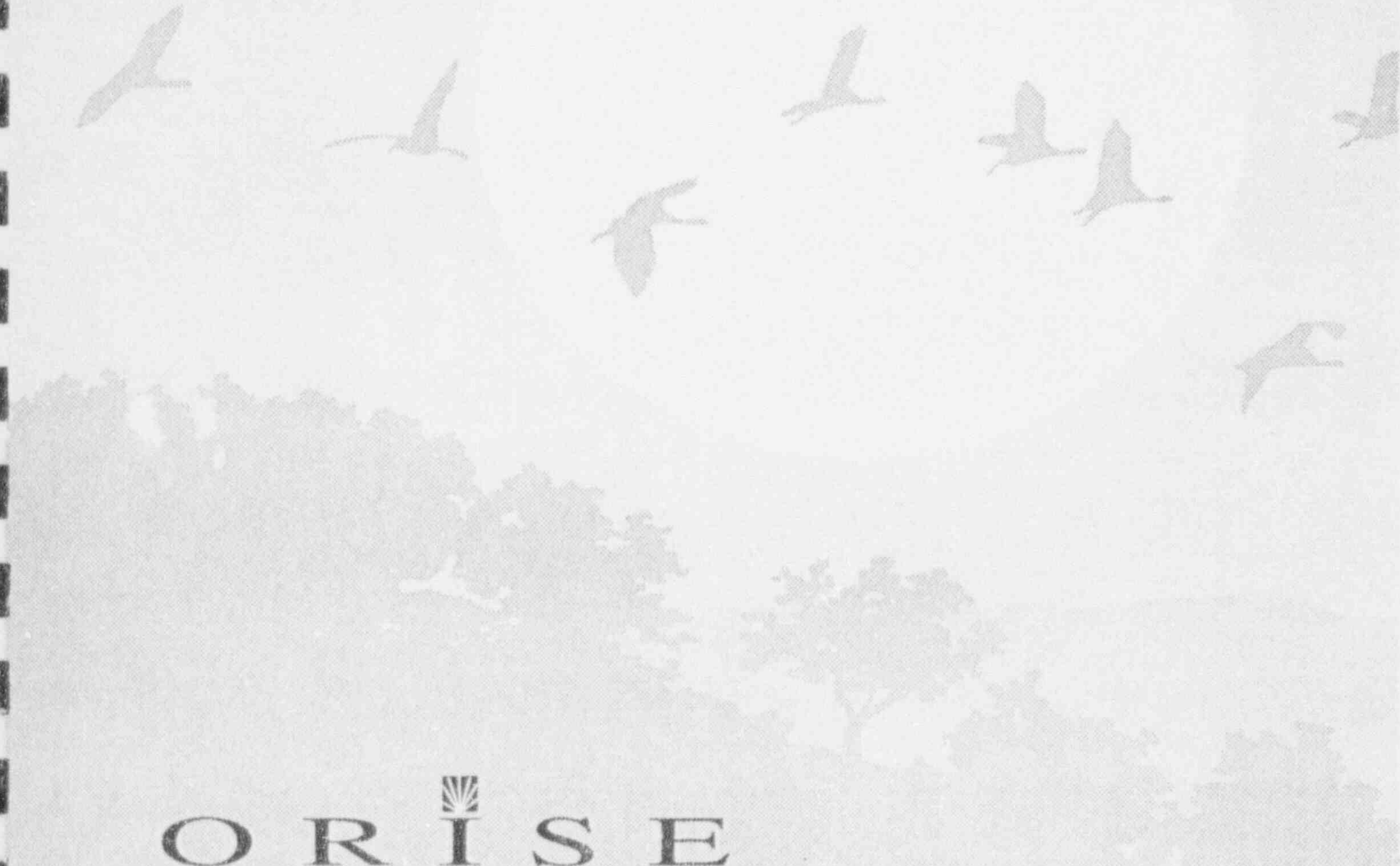

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FINAL REPORT

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BROOKHAVEN, NEW YORK

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ABBREVIATIONS AND ACRONYMS

ac	acre
ASME	American Society of Mechanical Engineers
cm^2	square centimeter
cpm	counts per minute
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^2	square feet
ha	hectare
GM	Geiger-Mueller
km	kilometer
LILCO	Long Island Lighting Company
LIPA	Long Island Power Authority
m^2	square meter
MDA	minimum detectable activity
mi	mile
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
QA	Quality Assurance
SNPS	Shoreham Nuclear Power Station
SE#	site exterior survey unit designation
SG#	site grounds survey unit designation
SU#	system survey unit designation
TB#	Turbine Building structural survey unit designation

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INTRODUCTION AND SITE HISTORY

The Long Island Lighting Company (LILCO) constructed a boiling water reactor, known as the Shoreham Nuclear Power Station (SNPS), which was designed to provide a gross electrical output of 849 Megawatts. Reactor criticality was achieved in February 1985. Low power testing, in accordance with U.S. Nuclear Regulatory Commission (NRC) License No. NPF-82 (NRC Docket File No. 50-322), which permitted reactor operations at levels not to exceed 5% of full power, commenced in July 1985. Reactor operations continued intermittently until January 1989, at which time power generating operations were terminated. The total operating history was equivalent to 2.03 effective full power days of fuel exposure. Irradiated fuel, which was a standard low enrichment (2 to 3% uranium-235) uranium fuel, was subsequently removed from the reactor vessel and placed into the spent fuel pool in August 1989.

Various reactor components, piping systems, and other equipment became radiologically contaminated as a result of reactor operation. The primary contaminants which have been identified during characterization studies include iron-55, cobalt-60, nickel-63, and smaller quantities of tritium, carbon-14, nickel-59, manganese-54, zinc-65, and europium-152.[1]

The Long Island Power Authority (LIPA) was established to decommission the facility and release the site for unrestricted use. LIPA's decommissioning plan was approved for implementation by the NRC in June 1992 and will include decontamination or removal of contaminated portions of the reactor and other plant systems and equipment. A major consideration of the decommissioning plan is to maintain the integrity when possible, of plant structures and systems. Activities involved with the decommissioning and termination surveys will be conducted over an approximate eighteen month period with the final phase being removal of the spent fuel from the site. The initial phase involved the termination survey of the internal components of the main turbine; which has since been followed by termination surveys of the remainder of the structures and systems located within the Turbine Building as well as the site grounds and building exteriors.

It is the policy of the NRC to perform confirmatory surveys of facilities that have undergone decommissioning and have requested NRC license termination. The NRC Headquarters' Division of Waste Management, formerly the Division of Low-Level Waste Management and Decommissioning, has requested that the Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) conduct confirmatory radiological surveys and related activities for the SNPS decommissioning project as the various decommissioning milestones are completed. The results of the confirmatory survey of the turbine internal components are the subject of a separate report.[2] This report describes the results of the confirmatory process which has been completed for the Turbine Building, site grounds, and building exteriors.

SITE DESCRIPTION

SNPS is located in the Town of Brookhaven, New York on the north shore of Long Island, approximately 80 km (50 mi) east of La Guardia Airport and the confluence of the East River and Long Island Sound (Figure 1). Reactor and supporting operations were conducted within a 32.4 ha (80 ac) portion of a larger 202 ha LILCO owned parcel of land that is bounded on the north by Long Island Sound, on the east by the Wading River Marshland, on the west by other LILCO property, and on the south by Route 25A. A cyclone fence encloses the 8 ha site secured area. Within this boundary are the buildings and grounds classified as the Restricted Area, also known as the power block, where radiological controls were necessary (Figure 2). Each of the buildings that are to be addressed during the confirmatory surveys are located here and are shown on Figure 2 as the Turbine Building, the Reactor Building, and the Rad Waste Building. Turbine Building construction is predominately of concrete and structural steel with a total floor space of 13,500 m^2 (145,000 ft^2) that is divided between three levels at elevations 15', 37' 6", and 63' (Figures 3 through 5). Surfaces and components within the building remain essentially intact following decommissioning activities. The systems and equipment housed include the turbine generator, main condenser, condensate system, feed water system, extraction heaters, part of the off-gas rad waste system, and drain sumps. Site grounds and site exteriors encompass those land areas and building exteriors and roof tops contained within the secured area.

Termination surveys have been performed in accordance with Draft NUREG/CR-5849.[3] LIPA has classified plant systems, building surfaces, and outside areas into two categories for survey, which are based on the potential for residual contamination. The two categories, referred to as affected or unaffected, are defined as follows: "affected areas are those areas which are potentially contaminated or have known contamination, or a system which circulated, stored or processed radioactive materials such that they could become contaminated, or experience, neutron activation, or where records indicated spills or other occurrences may have resulted in contamination; unaffected areas are those portions of the SNPS that are not expected to contain residual radioactivity." Area classification was determined by radiological use history, environmental monitoring activities, and the results of the previous characterization survey. Affected and unaffected areas are further subdivided into survey units. Survey units are categorized as structures (floors, walls, ceilings, and exterior surfaces of piping and equipment), plant systems (equipment and piping internals), and exterior areas (grounds and building exteriors). In addition, affected survey units also have sub-classifications as suspect or non-suspect, and may also be classified as alpha affected if involved with fuel handling or storage. For the Turbine Building, site grounds, and site exteriors, there were a total of 191 survey units addressed, of which, 135 were structures (including building exteriors), 48 were systems, and 8 were site grounds. Twenty-four of these survey units were classified by the licensee as affected.

OBJECTIVES

The objectives of the confirmatory activities were to provide independent document reviews, review and perform field observations of the LIPA procedures for embedded piping surveys, and develop radiological data for use by the NRC in evaluating the adequacy and accuracy of the licensee's procedures and termination survey results.

DOCUMENT REVIEW AND LIPA PROCEDURE SURVEILLANCE

ESSAP reviewed LIPA's termination survey procedures and the termination survey release records for those survey units selected for confirmatory survey.[4,5] Documents were reviewed for adequacy, accuracy, completeness, and consistency. In addition, ESSAP reviewed and initiated observational surveillance of the embedded piping procedures for appropriateness and consistency in field application.

PROCEDURES

During the period November 8 through 12, 1993, an ESSAP team visited the SNPS and performed independent visual inspections, measurements, and sampling of the Turbine Building, site grounds, and site exteriors. Surveys were performed in accordance with a survey plan submitted to and approved by the NRC.[6] ESSAP randomly selected 8 of the Turbine Building structural survey units, 3 of the system survey units and 1 each of the site ground and building exterior survey units for confirmatory surveys. In addition, the NRC site representatives, selected portions of 6 additional system survey units for confirmatory surveys. Survey unit designators are alpha-numeric with the first figures designating the type of unit, structural (building specific), system, grounds, or exteriors, followed by a three digit numeric reference. Subunits are given an additional two digit designation preceded by X. The survey units selected and the respective classification for each were:

Survey Unit	Survey Unit Name	Affected(A)/ Unaffected(U)	Structure/System/ Building Grounds
TB016	North Condenser Hallway	U	structure
TB017	West Condenser Bay	U	structure
TB031	Steam Seal Evaporator Room	A	structure
TB035	Truck Bay	A	structure
TB060	Chemistry Laboratory	A	structure
TB081	Re-Heater Area-East	A	structure
TB082	Re-Heater Area-West	A	structure

Survey Unit	Survey Unit Name	Affected(A)/ Unaffected(U)	Structure/System/ Building Grounds
TB089	Black Battery Charger Room	U	structure
SU005	Feed Water Control	A	system
SU014	Radwaste—Turbine Building Drain Piping System	A	system
SU014X03	Radwaste—Turbine Building Drain Piping System, Tank 12	A	system
SU024	Main Steam	A	system
SU025X2	Condensate and Feedwater, 15' Elevation Main Condenser	A	system
SU032	Lube Oil Sump Tank 91	U	system
SU034	Extraction Steam Valve 035C	A	system
SU046	Condensate Transfer and Storage	A	system
SU054X03	Low Conductivity and Salt Water Drains Influent Drain System Tank 186	A	system
SU071	Secondary Access Facility Ventilation System	U	system
SE002	Secured Area—North Buildings	U	structure
SG003	Secured Area West	U	bldg. grounds

Figures 3 through 6 indicate the structural and exterior survey units surveyed.

SURVEY PROCEDURES: INTERIOR

The following procedures apply to interior structural and system survey units.

Reference System

LIPA established the grid system used by ESSAP for referencing measurement and sampling locations. The grid size or reference interval established by LIPA for a given survey unit was dependent upon the survey unit classification (affected vs. unaffected) and surface (floor, lower wall, upper wall, ceiling, or equipment). Typically, floor and lower wall grid blocks were 1 m x 1 m. Upper surfaces, ceiling and equipment were either referenced to these grids or other prominent building features. Systems were referenced by a distance from a specific point, by drawings, or prominent components.

Surface Scans—Structural Units

Surface scans for alpha, beta, and gamma activity, were performed over 100% of floor, and lower wall surfaces, and up to 50% of equipment surfaces, within each structural survey unit. Additional scans were performed over portions of upper wall, ceiling, and/or system surfaces as well as locations, such as drains, where material may have settled or accumulated. Locations of elevated direct radiation detected by scans were marked for further investigation. Scans were performed using gas proportional, GM, and/or NaI detectors coupled to ratemeters or ratemeter-scalers with audible indicators.

Surface Activity Measurements

For each structural survey unit, ESSAP performed a minimum of thirty direct measurements for total beta surface activity at randomly selected locations. ESSAP also performed additional direct measurements at locations of elevated direct radiation detected by surface scans. At measurement locations where the average NRC surface contamination guideline was exceeded, the size of the contaminated area and the average activity in the contiguous 1 m^2 area was determined. Alpha surface activity measurements were not required as the s ected survey units were not classified as alpha affected and there was no alpha contamination identified by surface scans. Figures 7 through 14 show structural survey unit measurement locations. Measurements were performed using GM and/or gas proportional detectors coupled to ratemeter-scalers. A smear sample for determining removable activity level was collected from each direct measurement location.

Exposure Rate Measurements

Background exposure rate measurements were made at 10 locations within the Colt Building, which is of similar construction to the Turbine Building but did not have a history of radiological usage (Figure 15). Exposure rate measurements were performed at several direct measurement locations within structural survey units (Figures 7 through 14). All exposure rates were measured at 1 m above surfaces using a pressurized ionization chamber (PIC).

Systems

For multicomponent systems, ESSAP requested up to 5 randomly selected access points be opened to each of the system survey units selected for confirmatory survey. Beta and gamma surface scans were then performed within the accessible portions of the system followed by direct measurements and smear samples. Scans and direct measurements were performed using gas proportional, GM, and/or NaI detectors coupled to ratemeters or ratemeter-scalers. The total number of direct measurements performed was dependent upon component size and accessibility and ranged from 3 to 30 measurements (Figures 16 through 24).

Embedded Piping

Confirmation of the radiological status of embedded piping was accomplished primarily through surveillance of the methodologies and review of the procedures utilized. In addition, independent measurements were made within drain openings throughout the Turbine Building (Figure 25).

Comparative Measurements

LIPA was requested to perform direct measurements at 30 ESSAP direct measurement and exposure rate measurement locations. ESSAP instrumentation included GM and gas proportional detectors coupled to ratemeter-scalers for direct measurements and a PIC for exposure rate measurements. The LIPA instrumentation used, which ESSAP selected randomly, included HP-260 and 126 cm^2 GM detectors coupled to ratemeter-scalers for direct measurements and a micro-rem meter for exposure rate measurements. The LIPA micro-rem readings were converted to μR/h using the LIPA developed correlation factor of 3.06 + 1.07 (μrem).

SURVEY PROCEDURES: EXTERIOR

The following procedures apply to exterior site grounds and building exteriors.

Reference System

The grid systems established by LIPA on the site grounds and exterior building surfaces were used by ESSAP for reference.

Surface Scans

Site grounds, paved areas, and building exteriors were scanned for gamma activity, while paved areas and exterior building surfaces were also scanned for alpha and beta activity. Scans were performed using NaI, gas proportional, and/or GM detectors coupled to ratemeters or ratemeter-scalers with audible indicators.

Surface Activity Measurements

ESSAP performed 30 direct measurements for total beta activity on both the paved portions of SG003 and the surfaces of SE002 (Figures 26 through 28). Alpha direct measurements were not required. Direct measurements were made using gas proportional and/or GM detectors coupled to ratemeter-scalers. A smear sample for determining removable activity was collected from each direct measurement location.

Exposure Rate Measurements

Background exposure rate measurements were performed at 6 locations within 0.5 to 10 km of the site (Figure 29). Exposure rate measurements were also performed at each soil sampling location within SG003 (Figure 28). Exposure rates were measured at 1 m using a PIC.

Soil Sampling

Background soil samples were collected from 6 locations within 0.5 to 10 km of the SNPS (Figure 29). There were five soil samples collected from randomly selected locations within SG003 (Figure 28).

Confirmatory Analysis

Four soil samples and 1 septic tank sludge sample, collected by LIPA, were obtained for confirmatory analysis.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ESSAP's Oak Ridge laboratory for analysis and interpretation. Smears were analyzed for gross alpha and gross beta activity using a low background proportional counter. Soil and sludge samples were analyzed by solid state gamma spectrometry. The spectra were reviewed for Co-60 as well as any other identifiable photopeaks. Soil and sludge sample results were reported in units of pCi/g. Smear and direct measurement data were converted to units of dpm/100 cm^2. Direct measurements which exceeded background levels were corrected for Fe-55 contribution, which can not be adequately detected with field instrumentation. A correction factor of 1.2 was therefore applied to those surface activity measurements that exceeded background distribution levels. LIPA developed, and the NRC approved, the use of this correction factor based on the observed Co-60 to Fe-55 activity ratio identified in characterization samples.[4,7] Exposure rates were reported in μR/h. The 95% confidence level was calculated for surface activity and exposure rates for each survey unit selected for confirmation. A direct comparison of the ESSAP and LIPA survey unit results and individual soil sample results was performed. Additional information concerning major instrumentation, sampling equipment, and survey and analytical procedures is provided in Appendices A and B.

FINDINGS AND RESULTS

DOCUMENT REVIEW

ESSAP's review of the termination survey plan indicated that the document provided an adequate description of survey methodologies and general approaches. Comments were provided to the NRC in a January 12, 1993 correspondence.[8] ESSAP's review of the termination survey final report, and release records for those survey units selected for confirmatory survey, indicated that the survey plan had been appropriately followed with no significant deviations. Data were appropriately converted, tested, and presented. Comments which were identified may be summarized as follows:

- Several direct measurements exceeded the average activity guideline. There was no explanation of additional data provided in the report that would demonstrate compliance at these locations with the 1 m^2 average activity guideline. However, in the subsequent June 1994 issued Termination Survey Final Report, LIPA provided the 1 m^2 average activity data for each of these locations.[9] A review of this data indicated that the average activity levels were less than the guideline.

- The report would benefit from the incorporation of maps which provide an overall (assembled) system view indicating components, rather than only maps of individual components.

INTERIOR SURVEY UNITS

The results of the confirmatory survey of the interior survey units are discussed below.

Surface Scans

Alpha, beta, and gamma surface scans identified one small area of elevated direct beta radiation, measuring less than 15 cm^2 in area, on the floor of the Main Condensate Storage Tank (SU046) (Figure 22). A second, small area of elevated direct beta radiation was detected on the eleventh panel of the west wall, west outboard, B side of the Main Condenser (SU025) (Figure 19). All other surface scans were comparable to background levels.

Surface Activity Levels

The results of total and removable surface activity levels are summarized in Table 1. Total beta activity levels for the structural survey units ranged from -800 to 2,100 dpm/100 cm^2. Removable activity levels ranged from -1 to 6 dpm/100 cm^2 for alpha and -7 to 16 dpm/100 cm^2 for beta. Structural survey unit means ranged from -290 to 370 dpm/100 cm^2 and -1 to 1 dpm/100 cm^2 for total and removable beta activity respectively.

Total beta activity levels in the surveyed systems ranged from -910 to 5,800 dpm/100 cm^2. The removable activity levels were -1 to 4 dpm/100 cm^2 for alpha and -7 to 16 dpm/100 cm^2 for beta. The mean beta activity levels for systems ranged from -340 to 300 dpm/100 cm^2 for total activity and -1.0 to 1.8 dpm/100 cm^2 for removable activity.

Exposure Rates

Interior background exposure rates ranged from 4 to 5 μR/h and averaged 5 μR/h. Individual gross exposure rates within the Turbine Building ranged from 3 to 7 μR/h. The average gross exposure rates for all survey units ranged from 4 to 7 μR/h . Table 2 provides a summary of the interior exposure rates.

EXTERIOR SURVEY RESULTS

The following are the ~ sults of the confirmatory surveys of the exterior grounds and building exteriors.

Surface Scans

Surface scans of the exterior grounds, paved areas, and building exteriors did not identify any locations of elevated direct radiation.

Surface Activity Levels

Surface activity levels for the paved portions of SG003 and the building exteriors of SE002 are summarized in Table 1. Total beta activity levels ranged from -760 to 1300 dpm/100 cm^2 and removable activity ranged from -1 to 10 dpm/100 cm^2 for alpha and -7 to 7 dpm/100 cm^2 for beta. The mean beta activities were 4 dpm/100 cm^2 total and -1 dpm/100 cm^2 removable for SG003 and -270 dpm/100 cm^2 total and 0.8 dpm/100 cm^2 removable for SE002.

Exposure Rates

Background exposure rates ranged from 6 to 9 μR/h and averaged 8 μR/h. The gross exposure rates within SG003 ranged from 8 to 9 μR/h and averaged 8 μR/h. Table 3 provides a summary of exposure rate measurements.

Radionuclide Concentrations in Soils

Background soil sample concentration levels were less than 0.1 pCi/g of Co-60. The Co-60 concentration levels in the five samples collected from SG003 were less than or equal to 0.1 pCi/g. There were no other radionuclides identified, other than those occurring in nature.

ESSAP AND LIPA DATA COMPARISON

Comparative field measurements for surface activity and exposure rates were made in one survey unit using the various ESSAP and LIPA instrumentation. Most of the measurements collected from this unit were comparable to background; therefore, a meaningful statistical evaluation could not be performed. ESSAP intended to collect an additional 30 measurements from areas where positive residual activity, that is survey locations where the reported surface activity levels exceed both the background distribution and the minimum detectable activity of the instrumentation, was identified by LIPA. However, a sufficient quantity of these areas could not be located to provide a statistically significant number of measurement locations.

The Co-60 concentrations determined by ESSAP in each of the samples obtained for confirmatory analysis were less than 0.1 pCi/g as were the results of the LIPA analysis.

COMPARISON OF RESULTS WITH GUIDELINES

The confirmatory survey results were compared with both the data provided by LIPA and the generic and site-specific NRC guidelines for release to unrestricted use. The NRC's Regulatory Guide 1.86 provides the guidelines for acceptable surface contamination levels used to determine whether a licensed facility may be released to unrestricted use. These guidelines are summarized in Appendix C. The applicable guidelines are those for beta-gamma emitters of which Co-60 and Fe-55 are the primary contaminants at SNPS. The residual surface activity guidelines are:

Total Activity

5,000 dpm β-γ/100 cm^2, averaged over 1 m^2
15,000 dpm β-γ/100 cm^2, maximum in 100 cm^2

Removable Activity

1,000 dpm β-γ/100 cm^2

As previously discussed, the detection sensitivities of the field instruments are such that the residual Fe-55 activity can not be detected. Therefore, surface activity measurements were corrected for Fe-55 when appropriate. The mean surface activity level for each survey unit was calculated and the survey unit data tested at the 95% confidence level, relative to the guidelines, in accordance with Draft NUREG/CR-5849. These results are provided in Table 4 and 5.

A comparison of the ESSAP mean activity levels to the LIPA mean activity level showed that in each instance the ESSAP mean was statistically less than or equal to the respective mean determined by LIPA for 16 out of the 19 confirmatory survey units. The ESSAP mean was greater than the LIPA mean for survey units TB017, TB089, and SU054X03. For TB017 and SU054X3, the activity levels were indistinguishable from background and the differences are therefore not considered significant. The differences in means observed for TB089 were the result of a higher background level associated with this room. LIPA established a separate background, 189 cpm versus the 129 cpm background used in data conversions for other survey units, for this room while ESSAP did not. The ESSAP decision was based on there being no indications of contamination (surface scans did not identify any locations of direct radiation and all measurements were less than MDA) and the effect of using a different background was inconsequential, relative to the overall status of the survey unit. Surface activity levels within each survey unit satisfied the guidelines at the 95% confidence level. There were no measurements which exceeded the maximum activity guideline. Two direct measurements exceeded the 5,000 dpm/100 cm^2 average guideline, both of which were identified within affected systems. An activity level of 5,100 dpm/100 cm^2 was detected at one location within the Main Condenser (SU025). The elevated activity was confined to an area of approximately 100 cm^2. The average activity within the contiguous 1 m^2 area was 2,800 dpm/100 cm^2. The second location measured 5,800 dpm/100 cm^2 and was confined to an area of less than 15 cm^2 on the floor of the Main Condensate Transfer and Storage Tank (SU046). The average activity in the surrounding 1 m^2 area was 1,300 dpm/100 cm^2. All remaining total and removable activity levels were below guideline values.

Exposure rates were compared with those obtained by LIPA, and tested at the 95% confidence level, relative to the 5 μR/h above background guideline currently being used by the NRC (Table 4)[9]. The interior and exterior exposure rates were both comparable to the respective background exposure rate level and confirmed the findings presented by LIPA.

Soil concentrations in the samples collected by ESSAP and the LIPA samples, obtained for confirmatory analysis, were compared with the 8 pCi/g site-specific limit for SNPS produced radionuclides.[4,5] The Co-60 level in all samples was less than or equal to 0.1 pCi/g.

SUMMARY

ESSAP performed confirmatory activities of the Turbine Building, site grounds, and site exteriors at the Shoreham Nuclear Power Station in Brookhaven, New York. Confirmatory activities included document reviews, and during the period November 8 through 12, 1993, independent surface scans, surface activity measurements, exposure rate measurements, and soil sampling, were performed.

The survey results confirm the results of the LIPA termination surveys. These findings indicate that surface activity levels, exposure rates, and soil concentration levels were below the NRC and site-specific guidelines for release to unrestricted use. Statistical tests of data sets further support the conclusion that each survey unit satisfies the guidelines at the 95% confidence level.

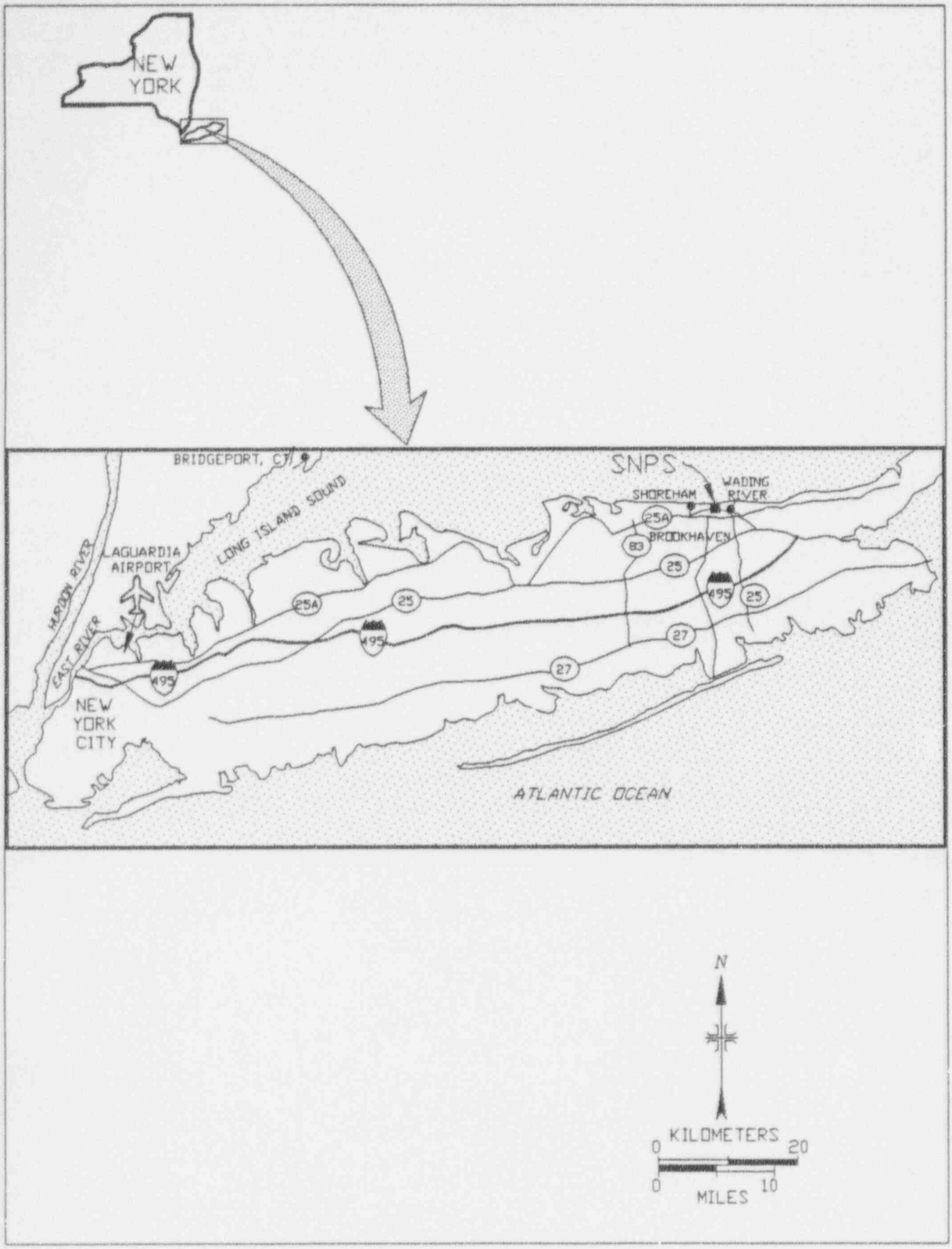


FIGURE 1: Location of the Shoreham Nuclear Power Station

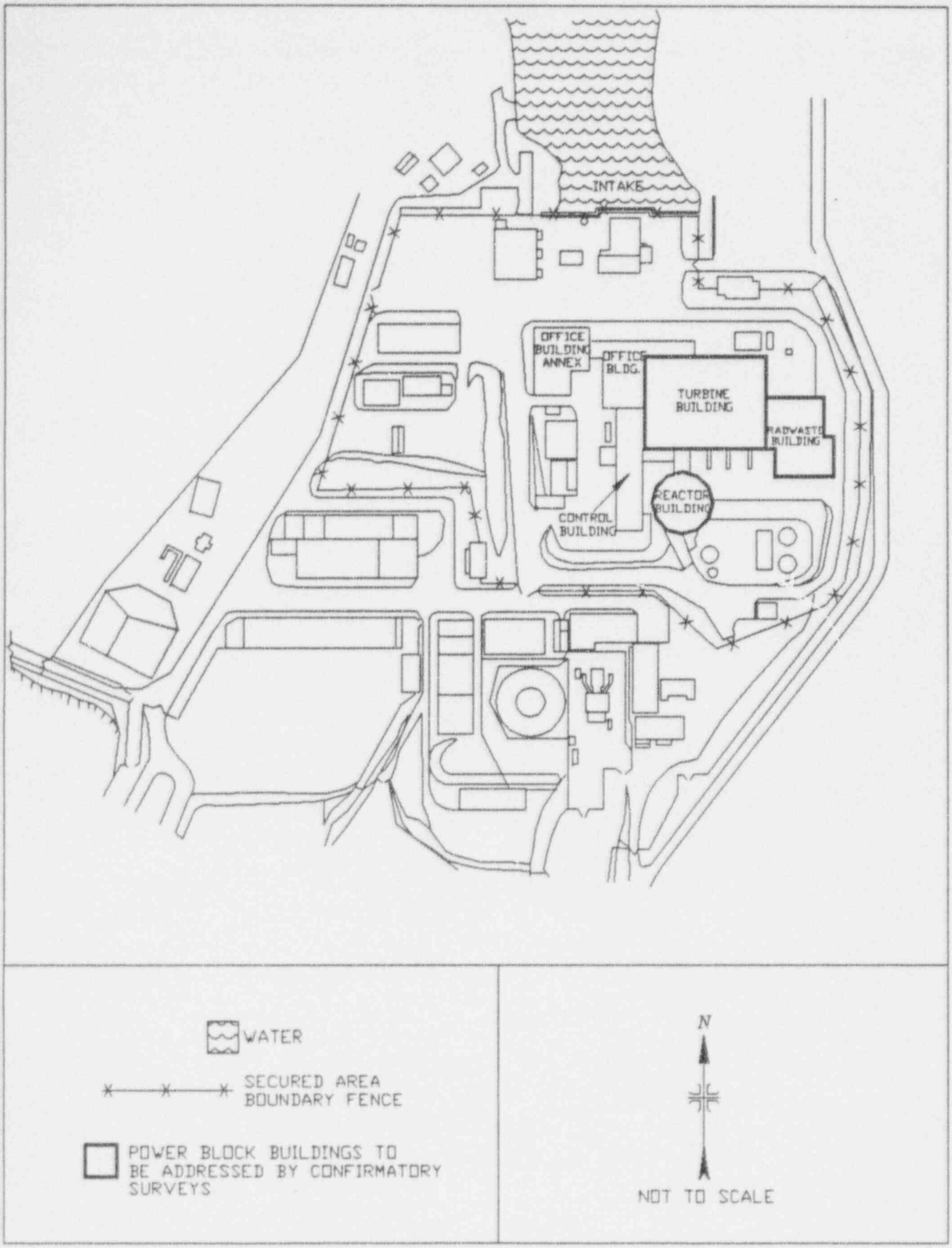


FIGURE 2: Plot Plan of the Shoreham Nuclear Power Station

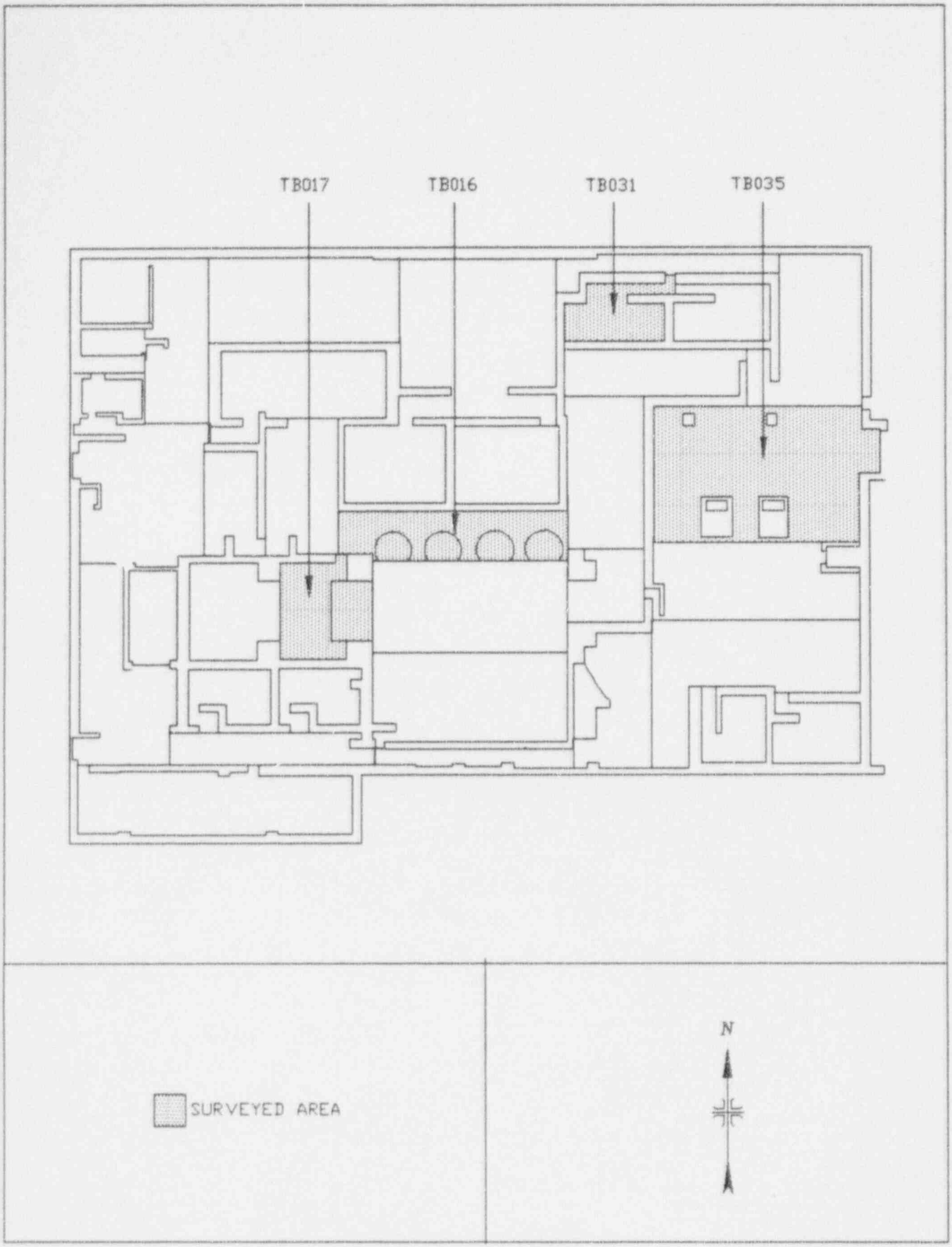


FIGURE 3: Turbine Building, Elevation 15' - Floor Plan and Areas Surveyed

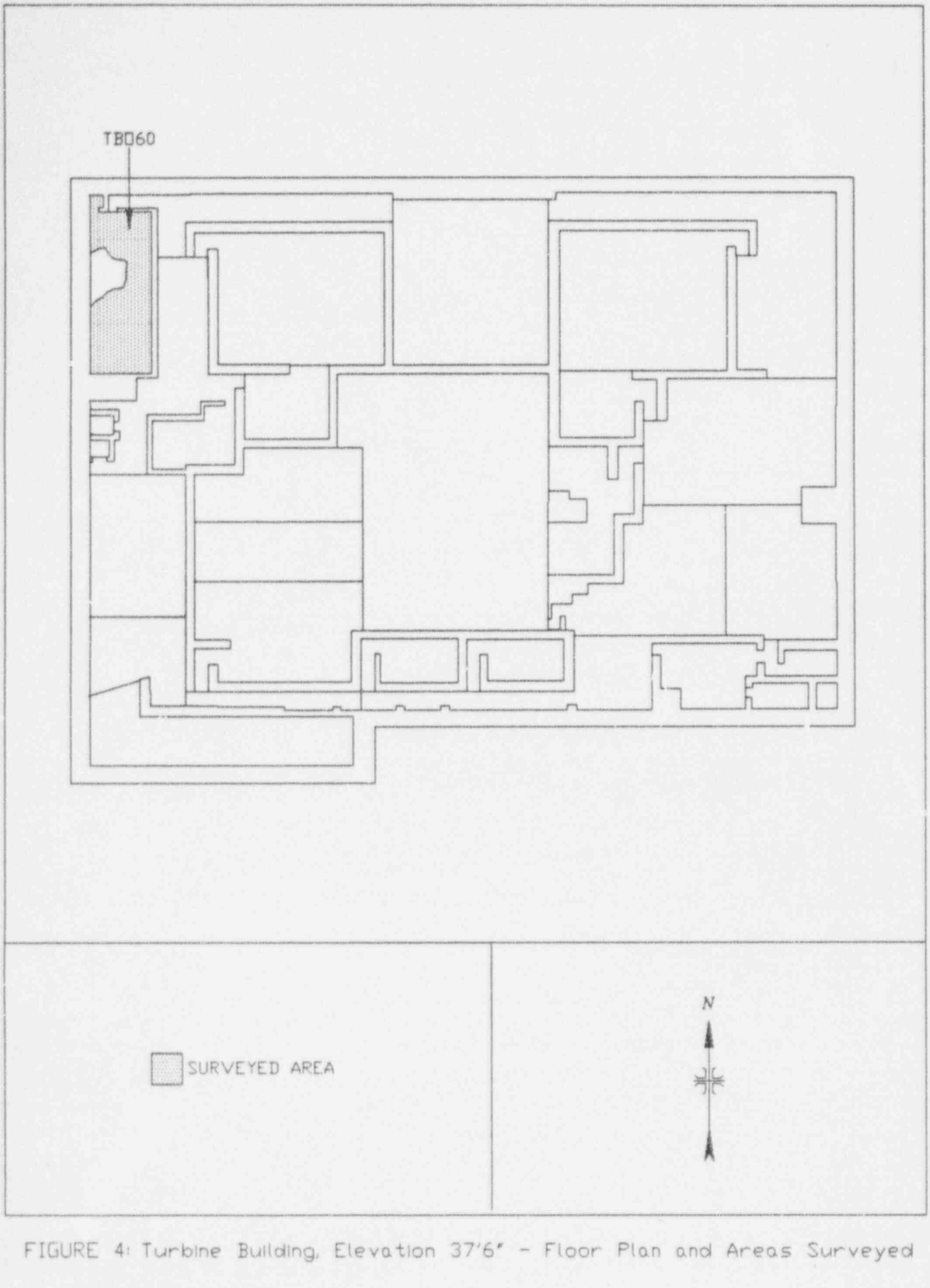


FIGURE 4: Turbine Building, Elevation 37'6" - Floor Plan and Areas Surveyed

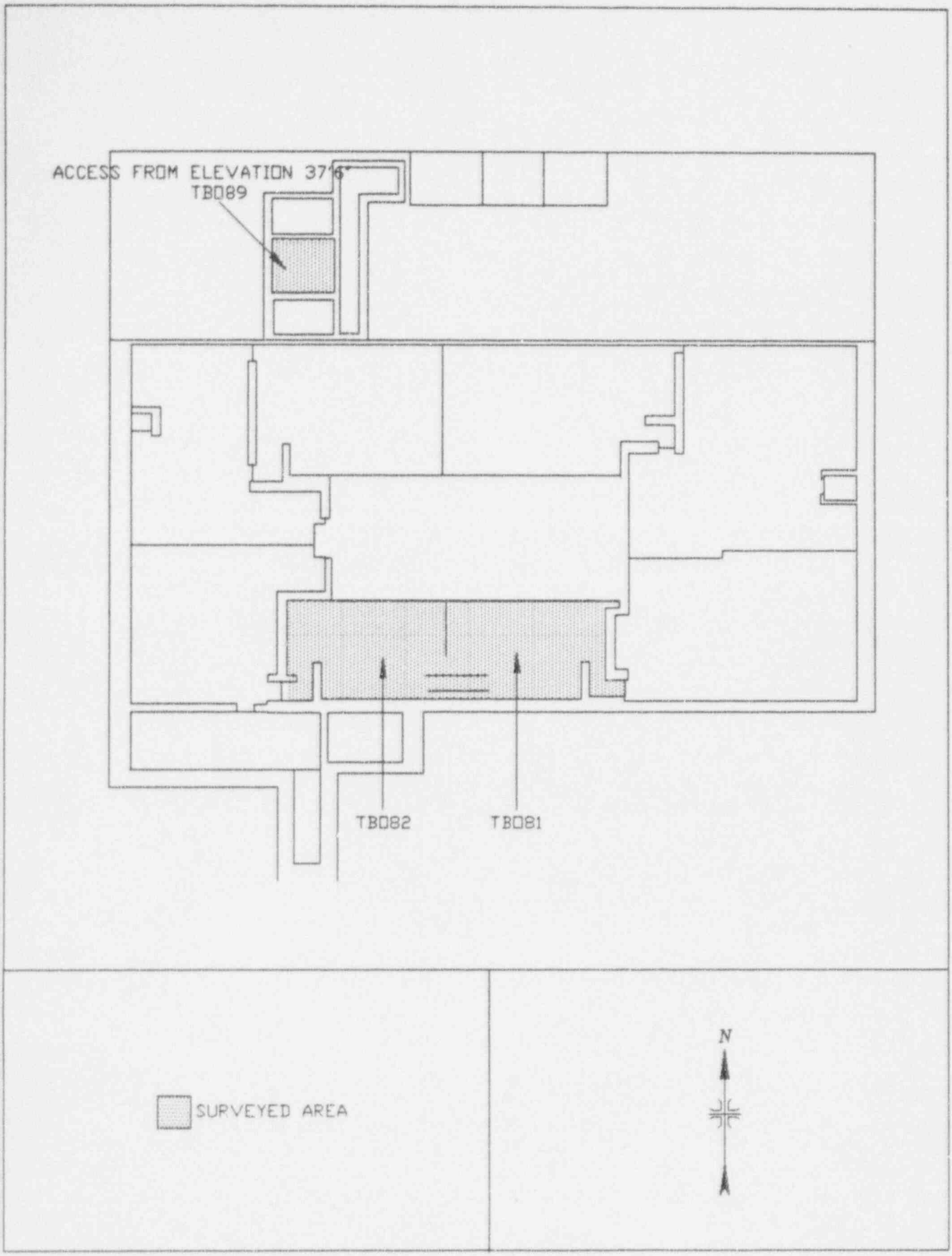


FIGURE 5: Turbine Building, Elevation 63' - Floor Plan and Areas Surveyed

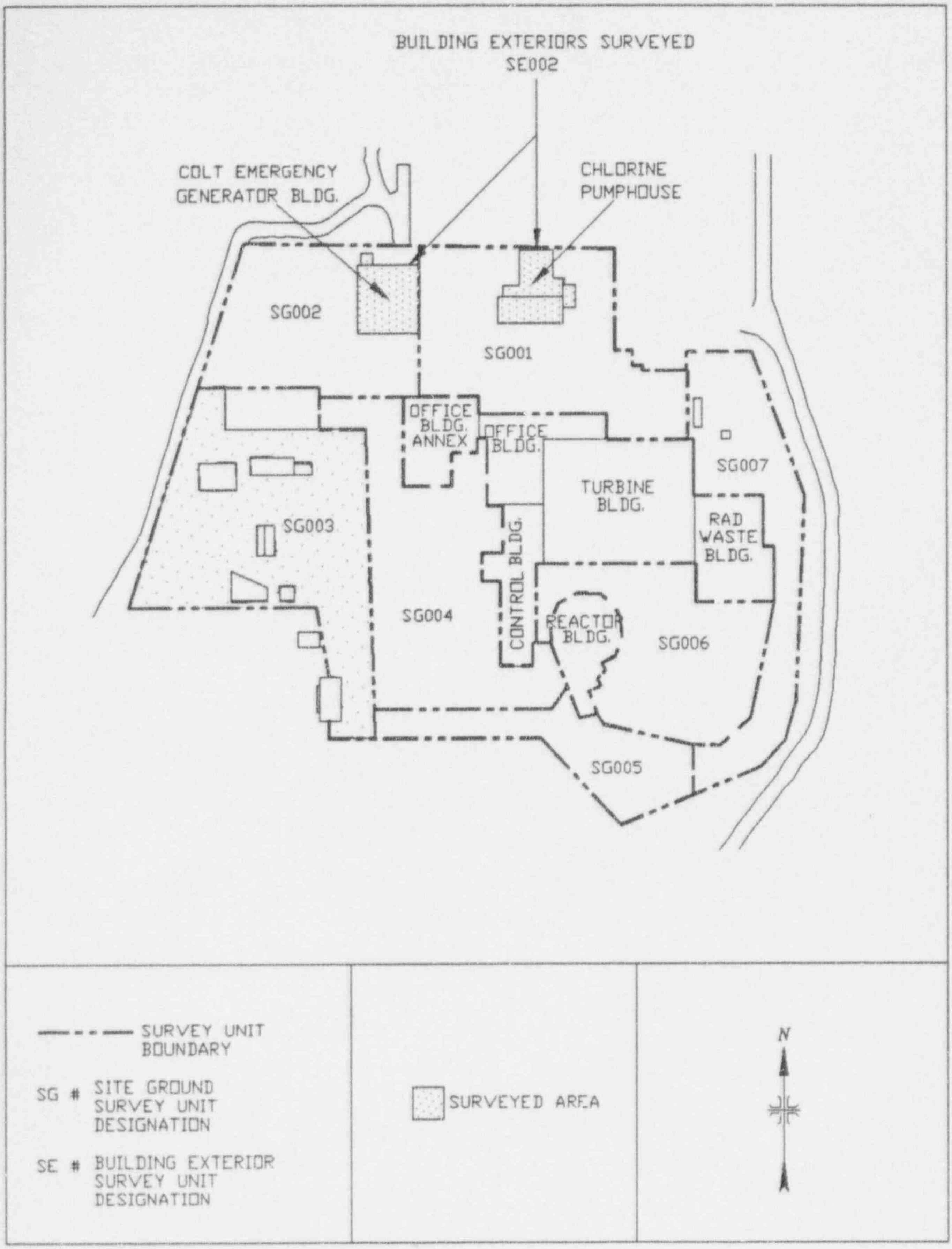


FIGURE 6: Shoreham Nuclear Power Station, Restricted Area - Exterior Areas Surveyed

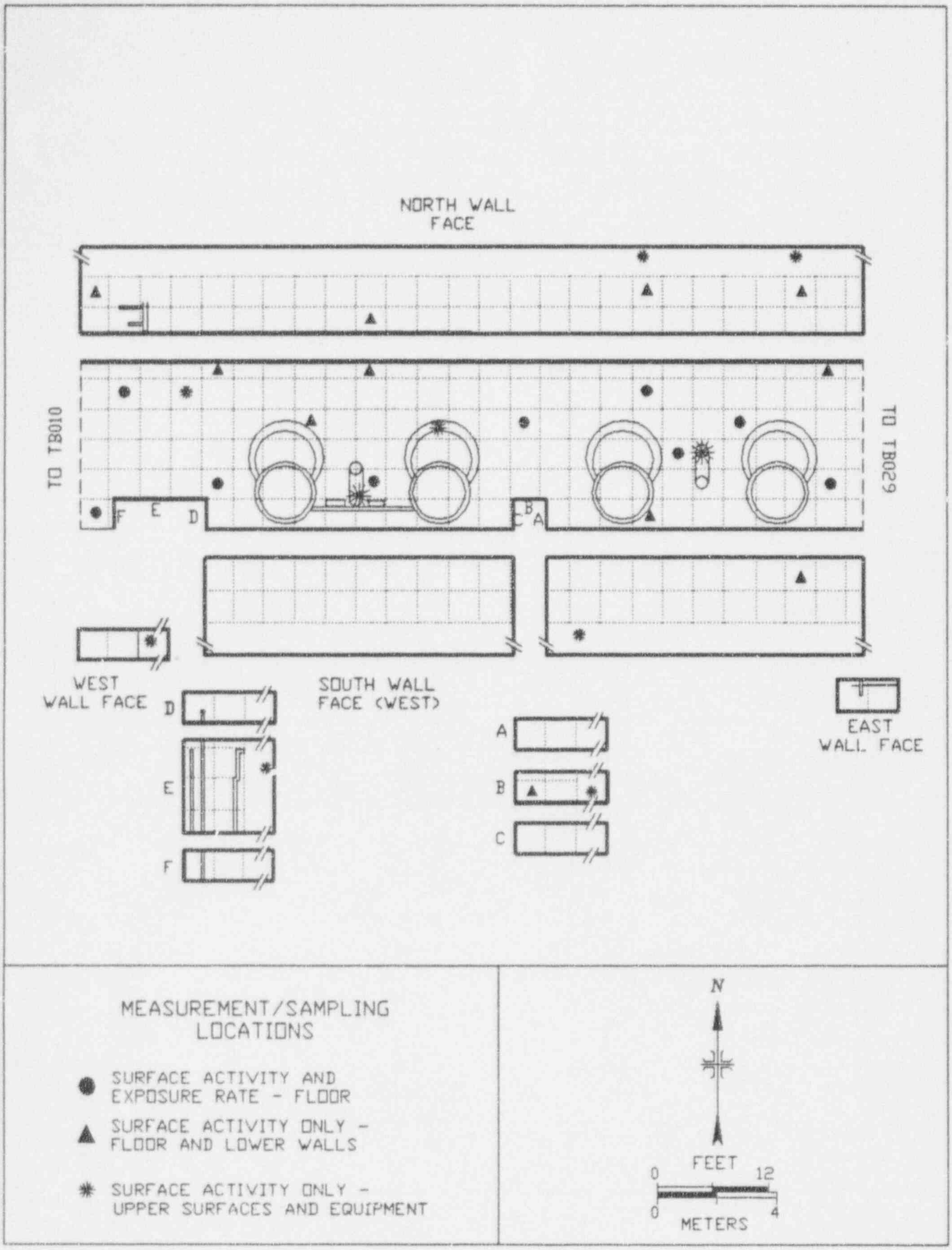


FIGURE 7: Turbine Building, North Condenser Hallway (TB016) - Measurement and Sampling Locations

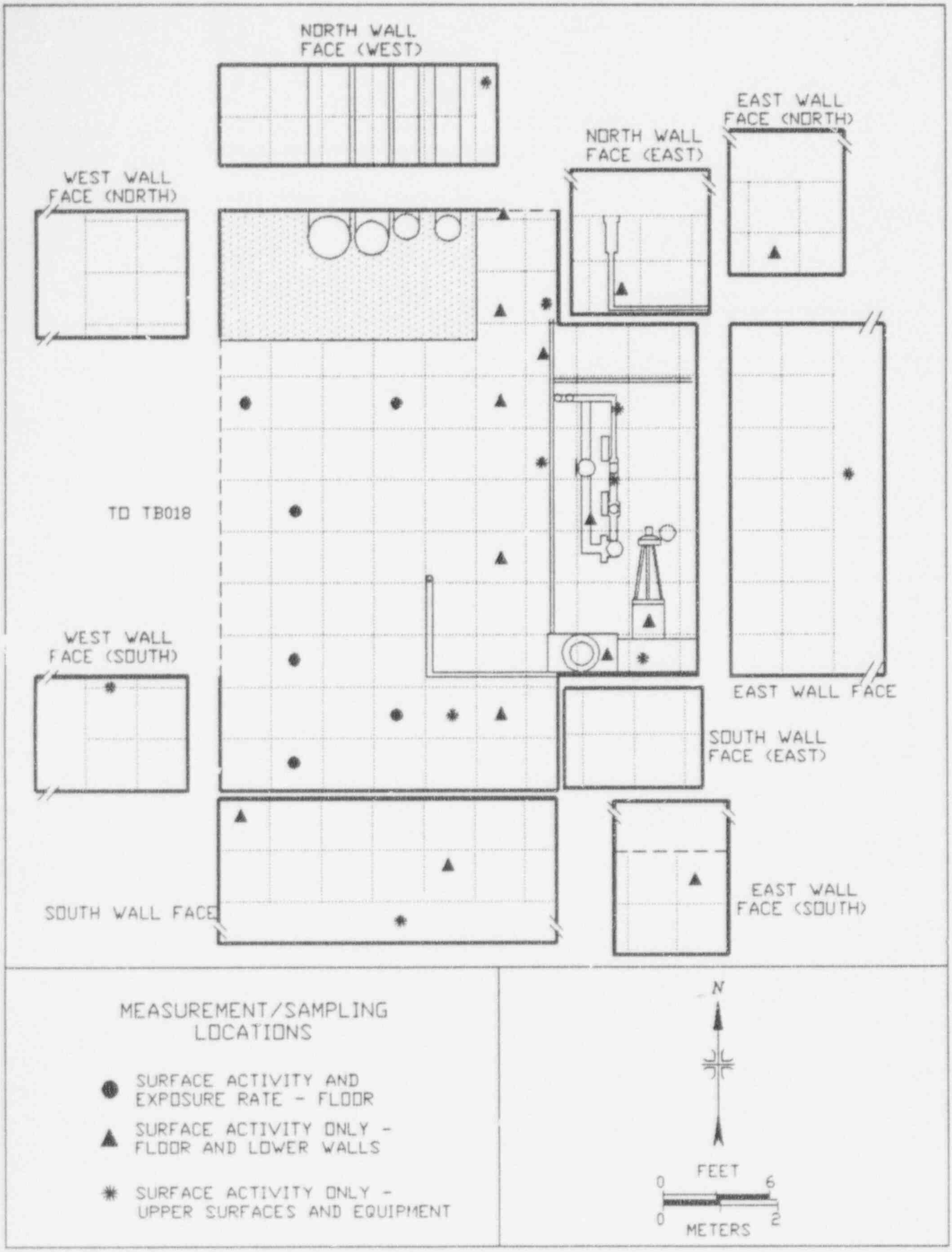


FIGURE 8: Turbine Building, West Condenser Bay (TB017) - Measurement and Sampling Locations

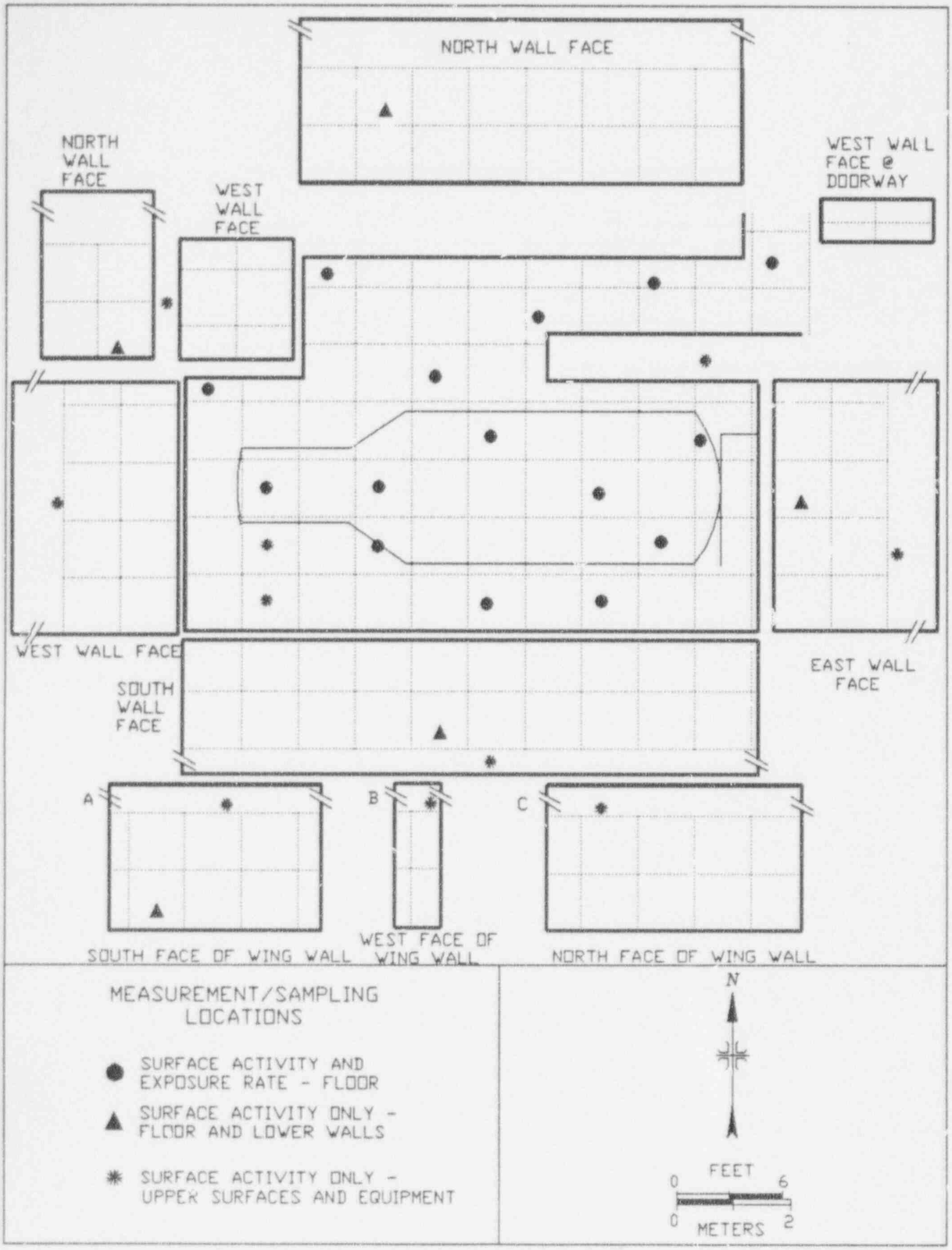


FIGURE 9: Turbine Building, Steam Seal Evaporator Room (TB031) - Measurement and Sampling Locations

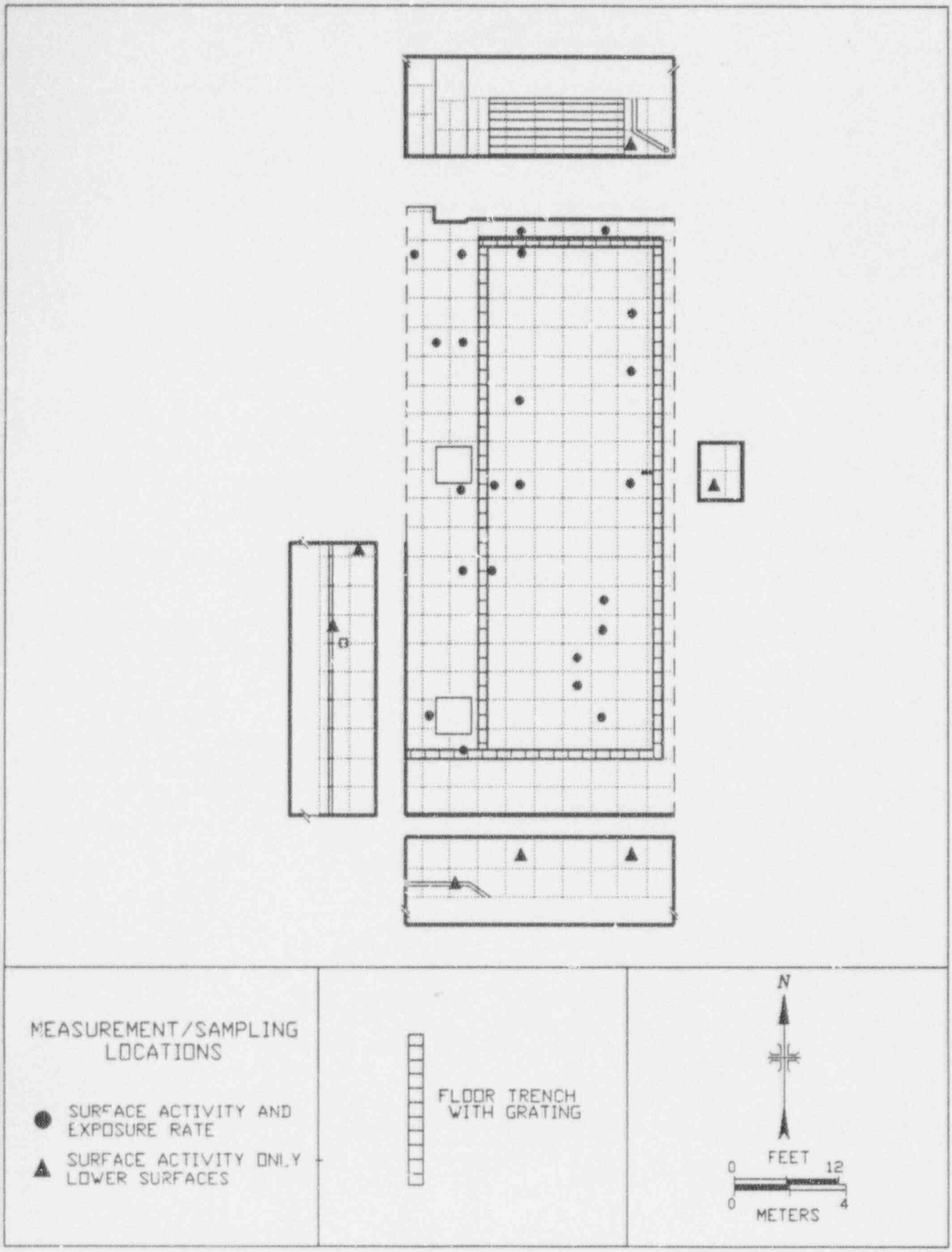


FIGURE 10: Turbine Building, Truck Bay (TB035) - Measurement and Sampling Locations

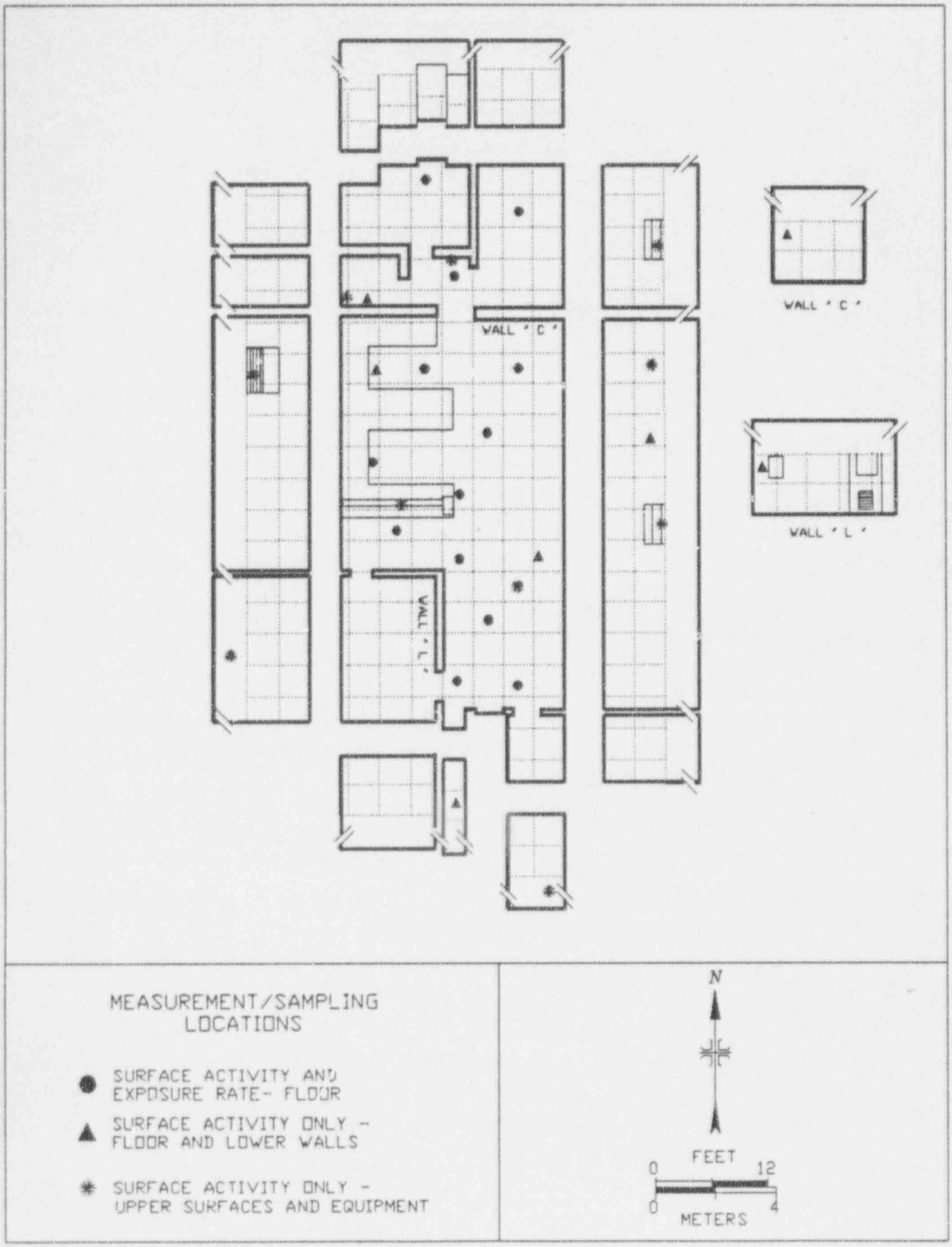


FIGURE 11: Turbine Building, Chemistry Laboratory (TB060) - Measurement and Sampling Locations

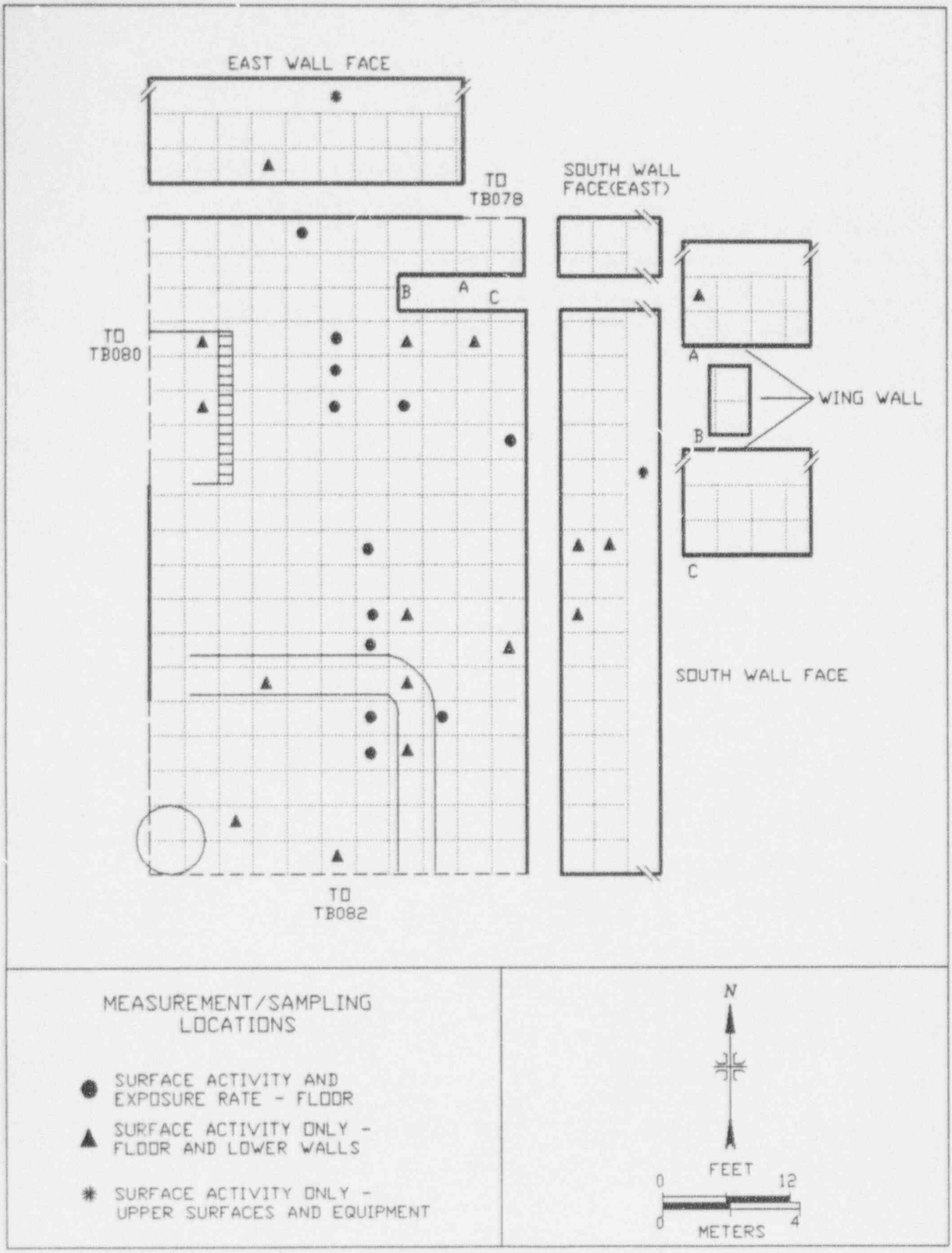


FIGURE 12: Turbine Building, Re-heater Area, East (TB081) - Measurement and Sampling Locations

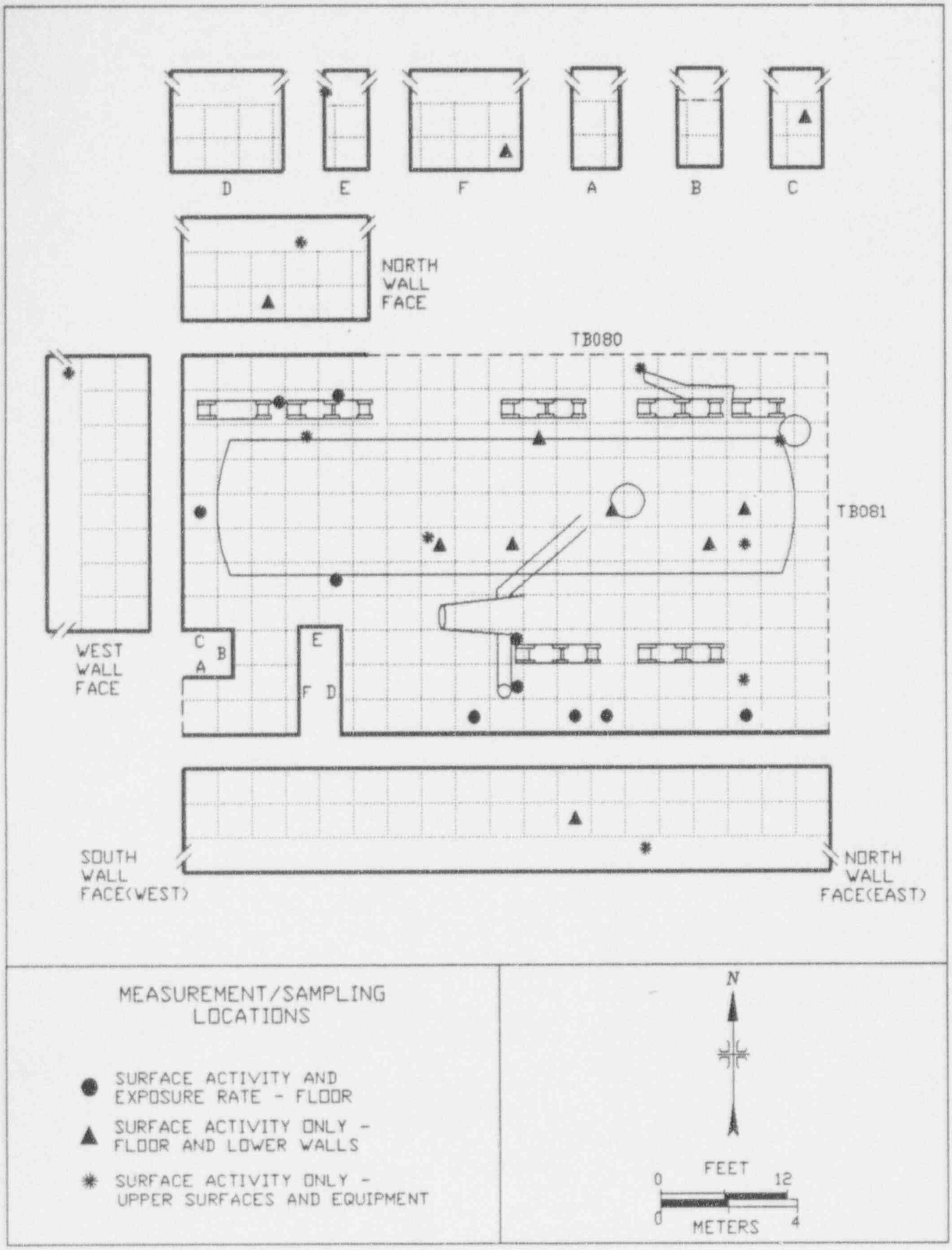


FIGURE 13: Turbine Building, Re-heater Area, West (TB082) - Measurement and Sampling Locations

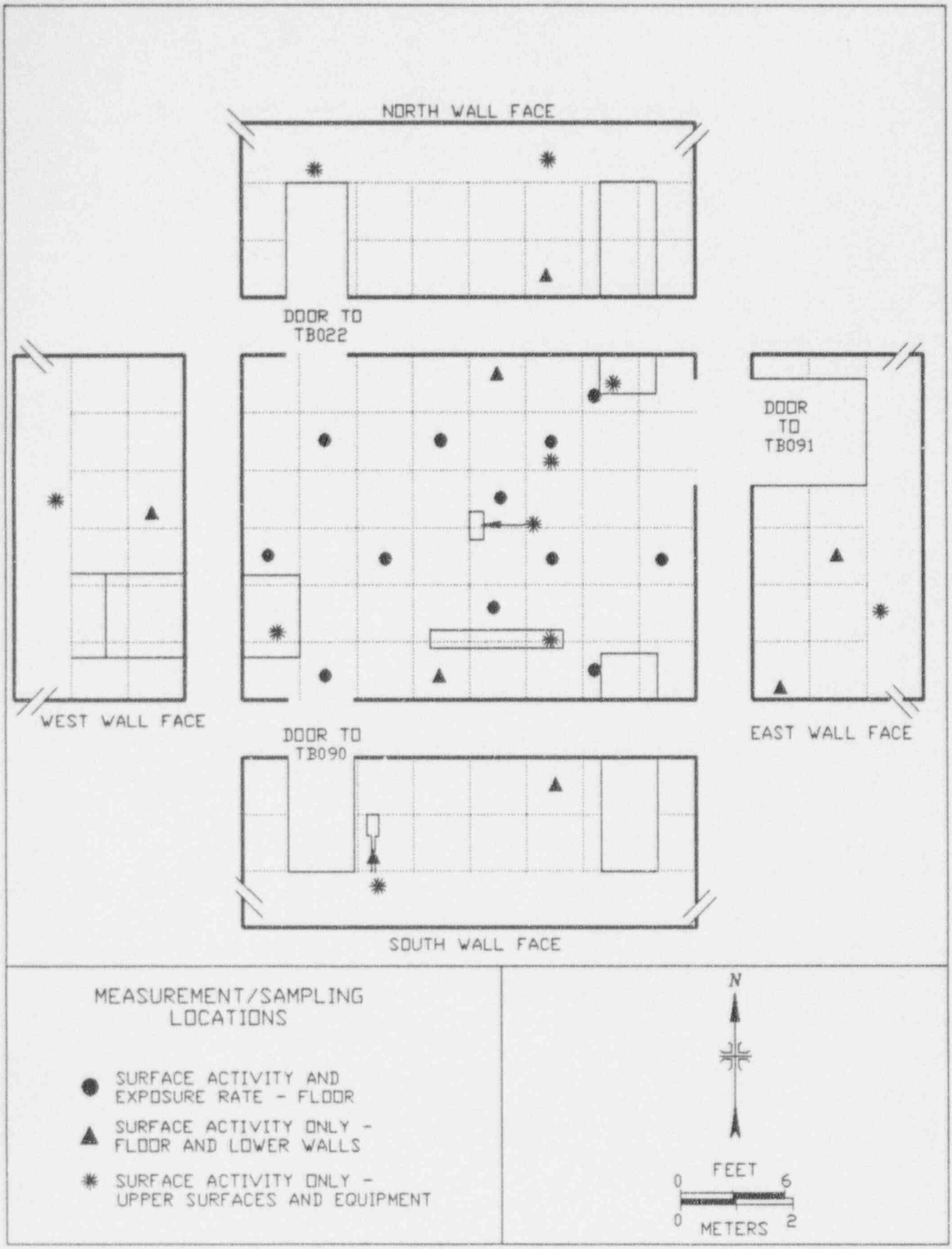


FIGURE 14: Turbine Building, Black Battery Charger Room (TB089) - Measurement and Sampling Locations

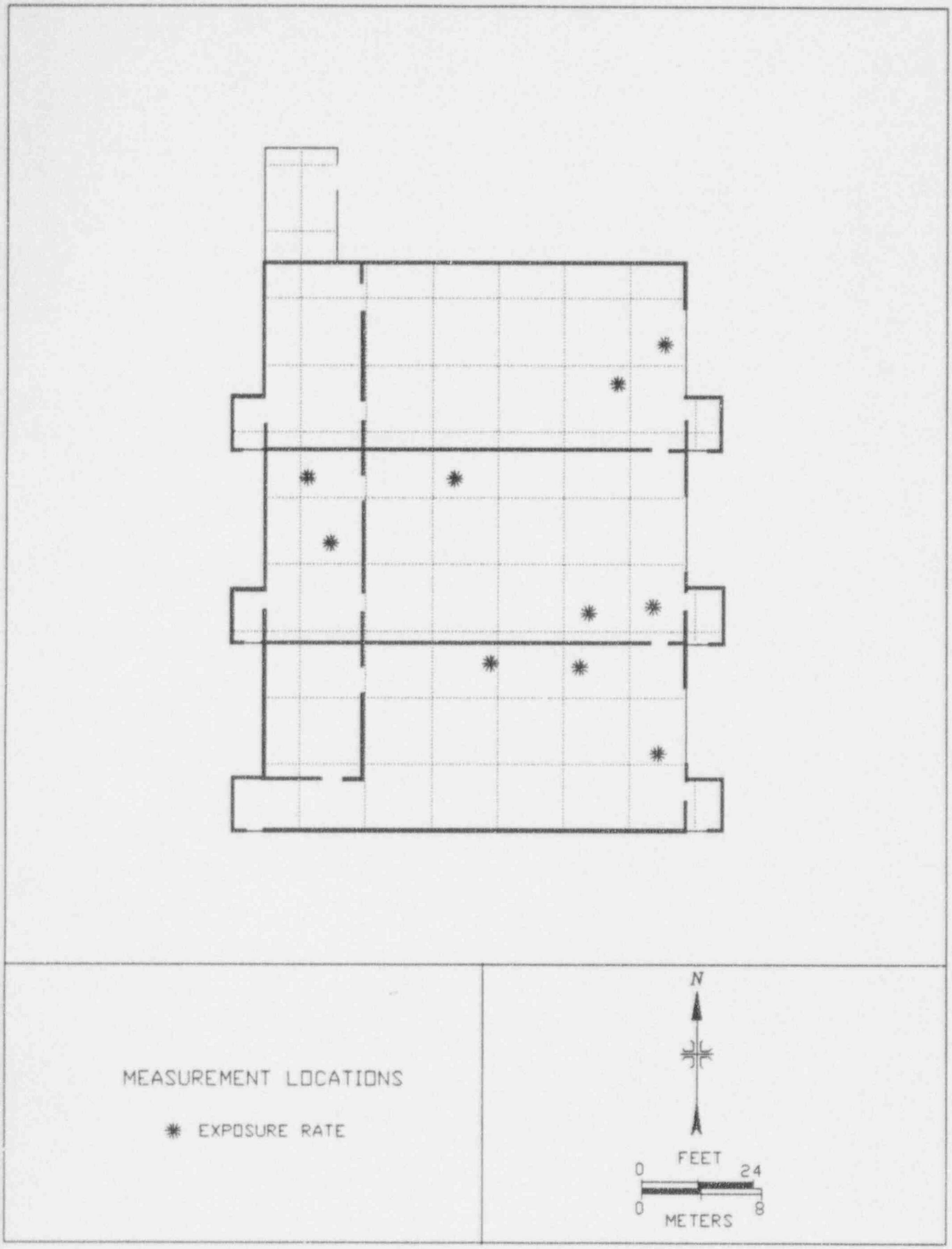
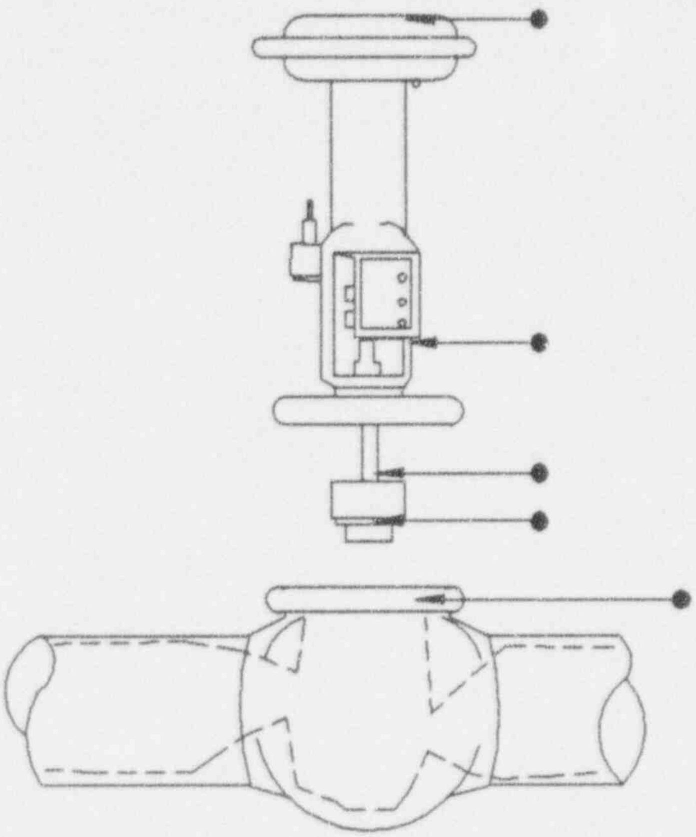


FIGURE 15: Colt Emergency Diesel Generator Building - Background Measurement Locations



MEASUREMENT/SAMPLING
LOCATIONS

● SURFACE ACTIVITY

NOT TO SCALE

FIGURE 16: Turbine Building, Feedwater Control (SU005) - Measurement and Sampling Locations

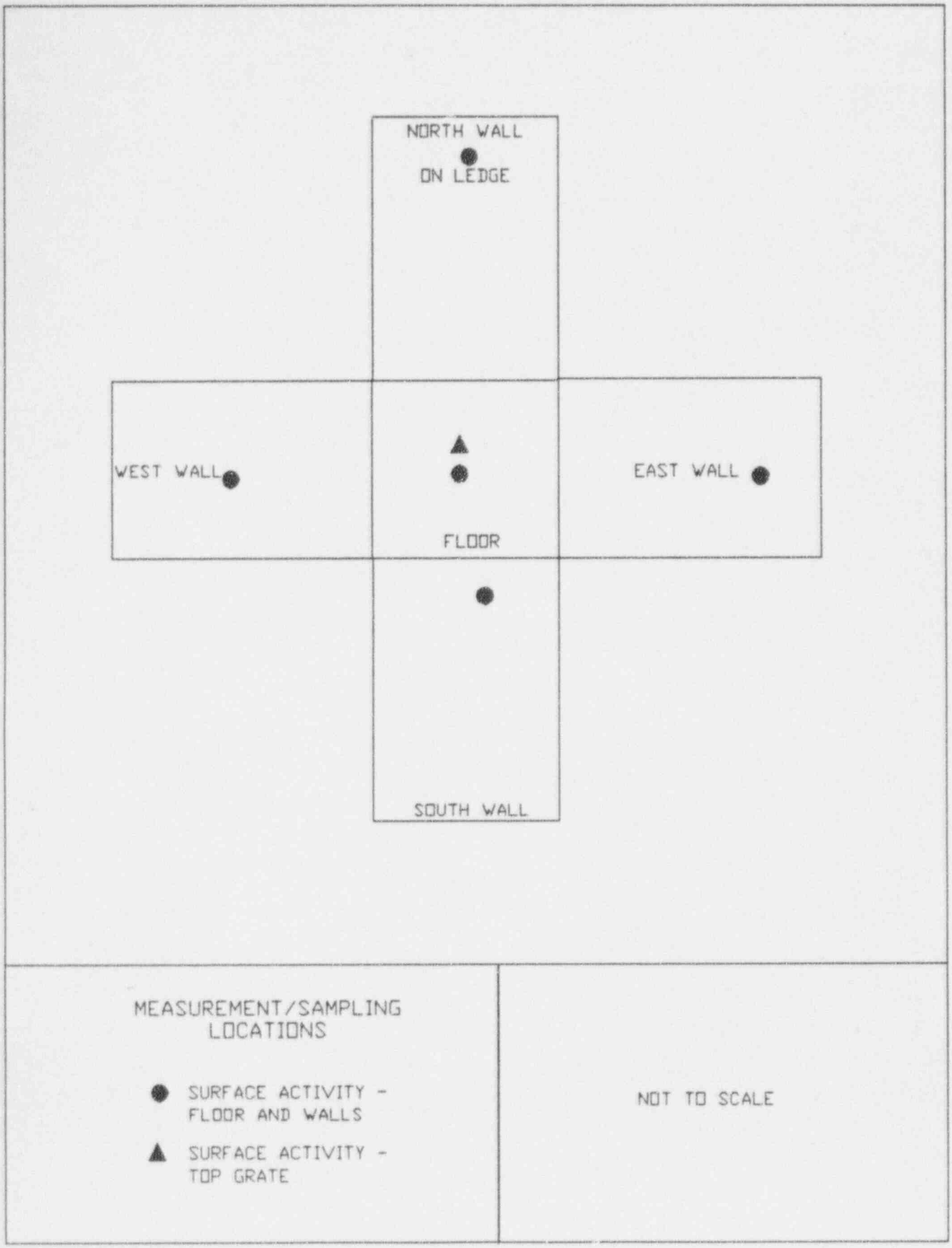


FIGURE 17: Turbine Building, Radwaste Interior Drain Pipe System, Tank 12 (SU014X03) - Measurement and Sampling Locations

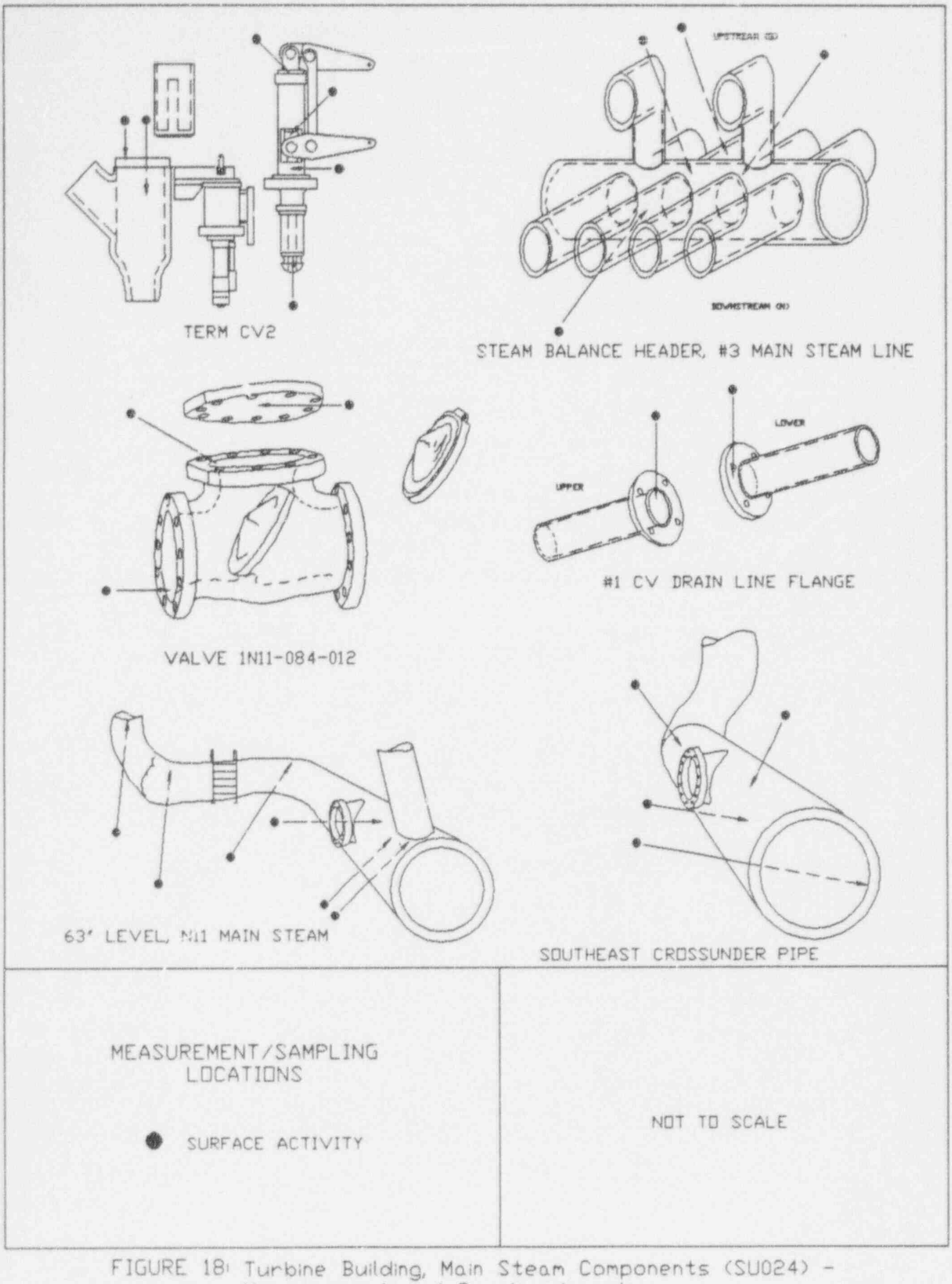


FIGURE 18 Turbine Building, Main Steam Components (SU024) - Measurement and Sampling Locations

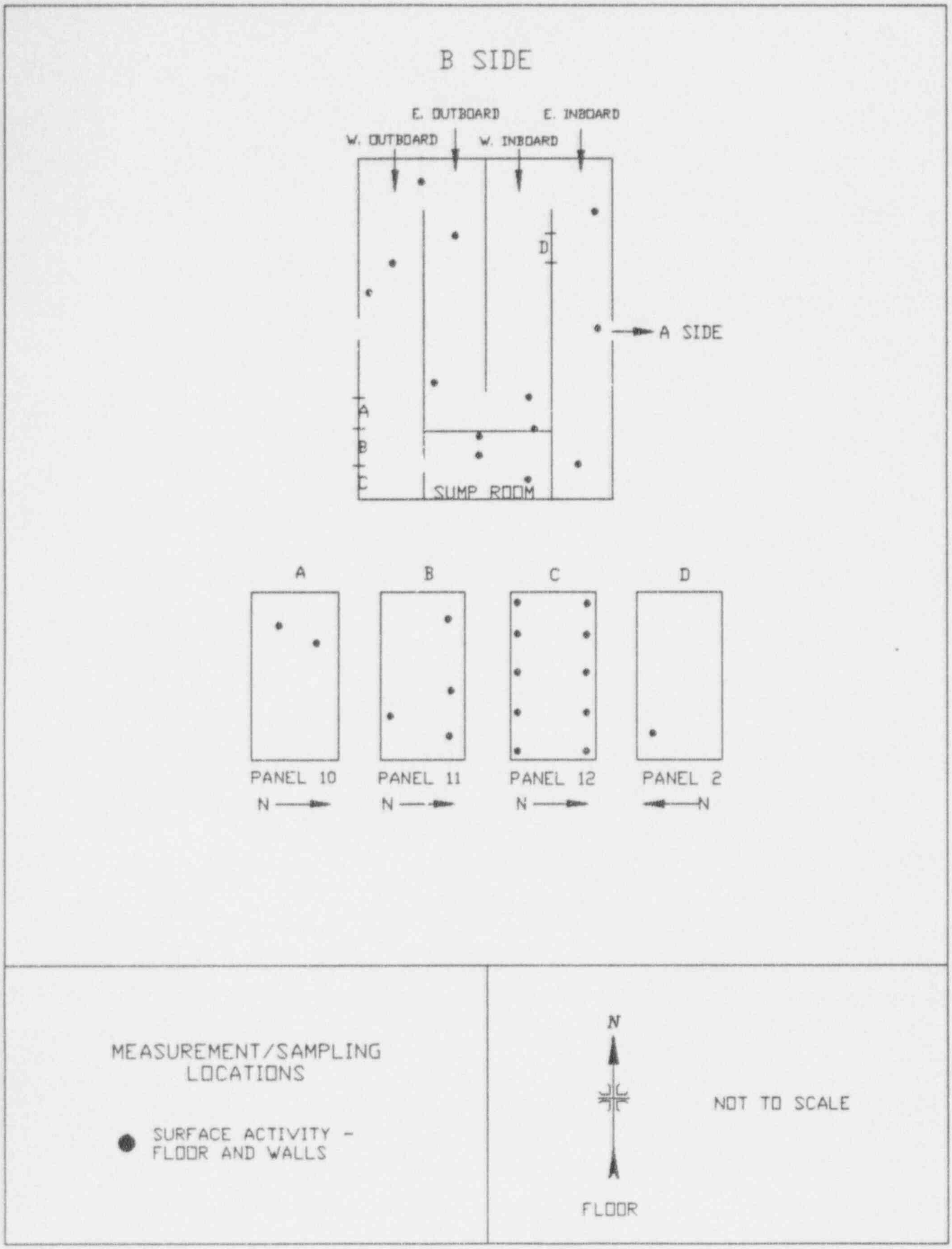


FIGURE 19: Turbine Building, 15′ Elevation, Condensate and Feedwater, Main Condenser (SU025X02) - Measurement and Sampling Locations

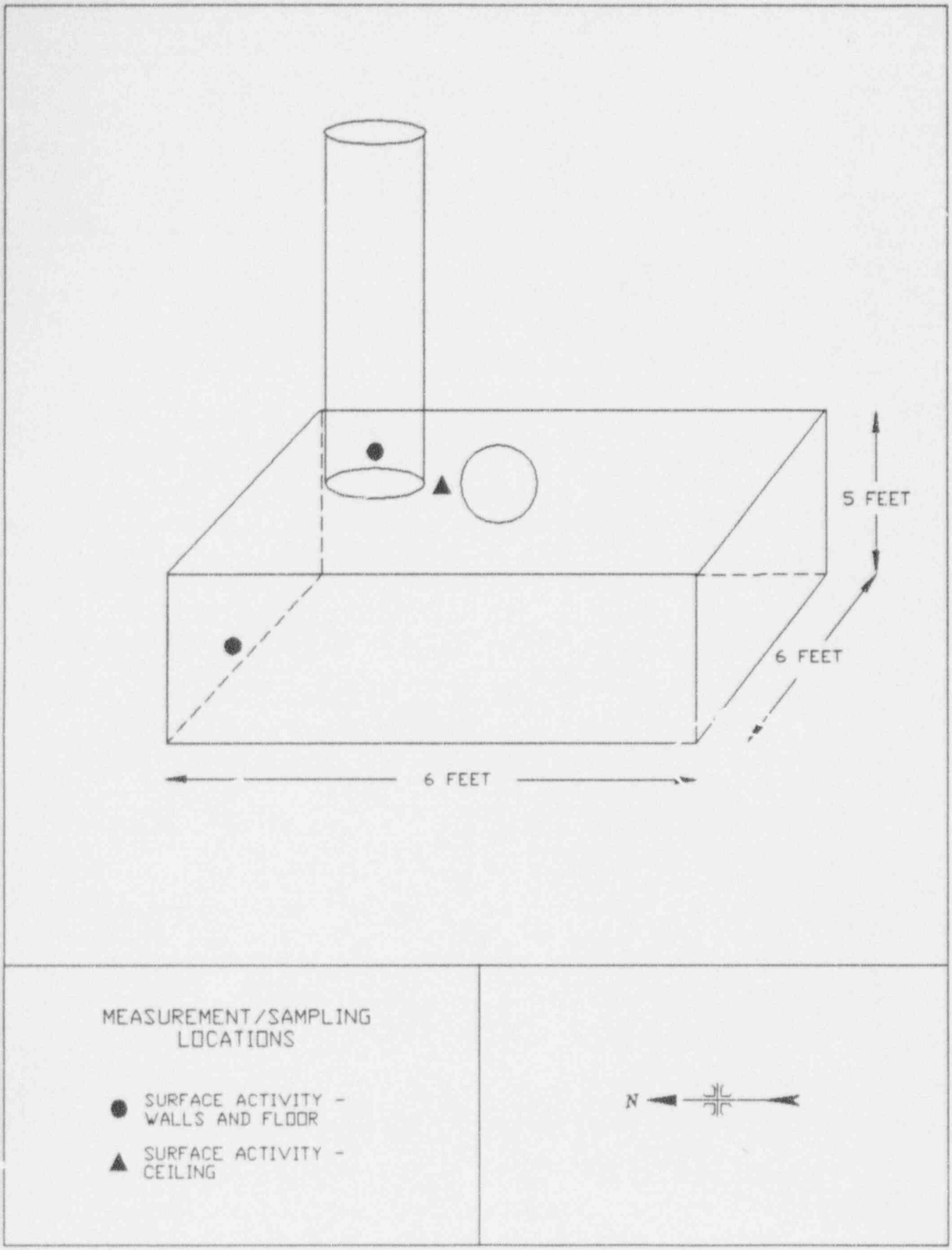
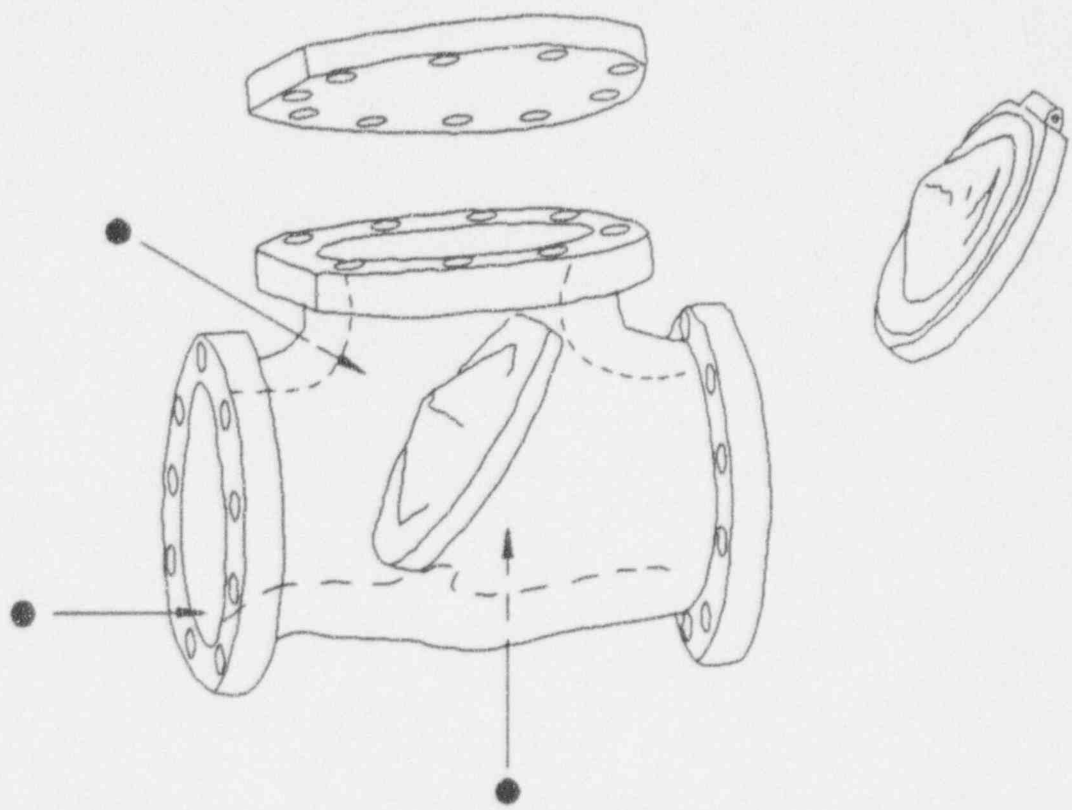


FIGURE 20: Turbine Building, Lube Oil Sump Tank 91 (SU032) - Measurement and Sampling Locations



MEASUREMENT/SAMPLING LOCATIONS

● SURFACE ACTIVITY

NOT TO SCALE

FIGURE 21: Turbine Building, Extraction Steam Valve 035C (SU034) - Measurement and Sampling Locations

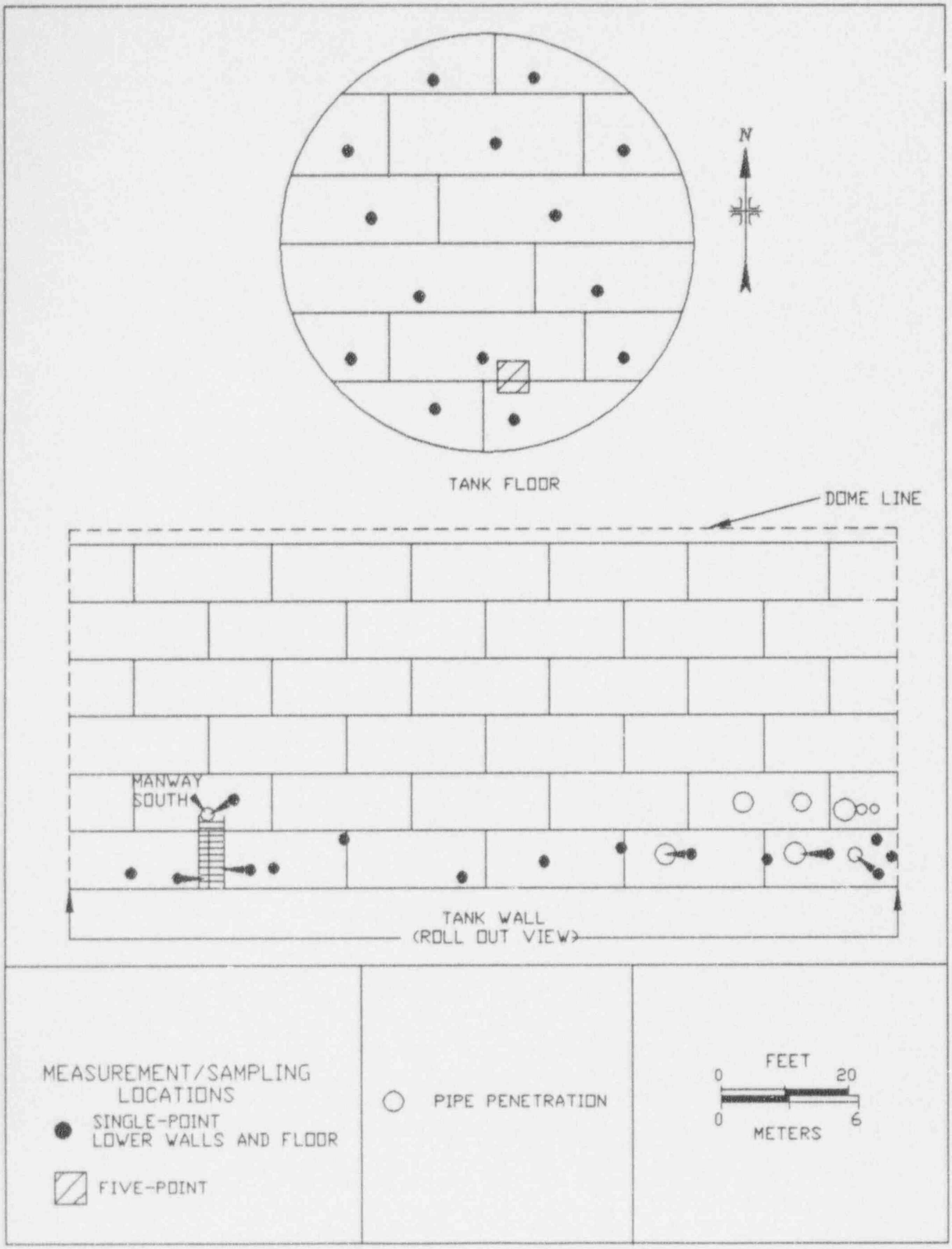


FIGURE 22: Condensate Transfer and Storage, Condensate Storage Tank (SU046X02) - Measurement and Sampling Locations

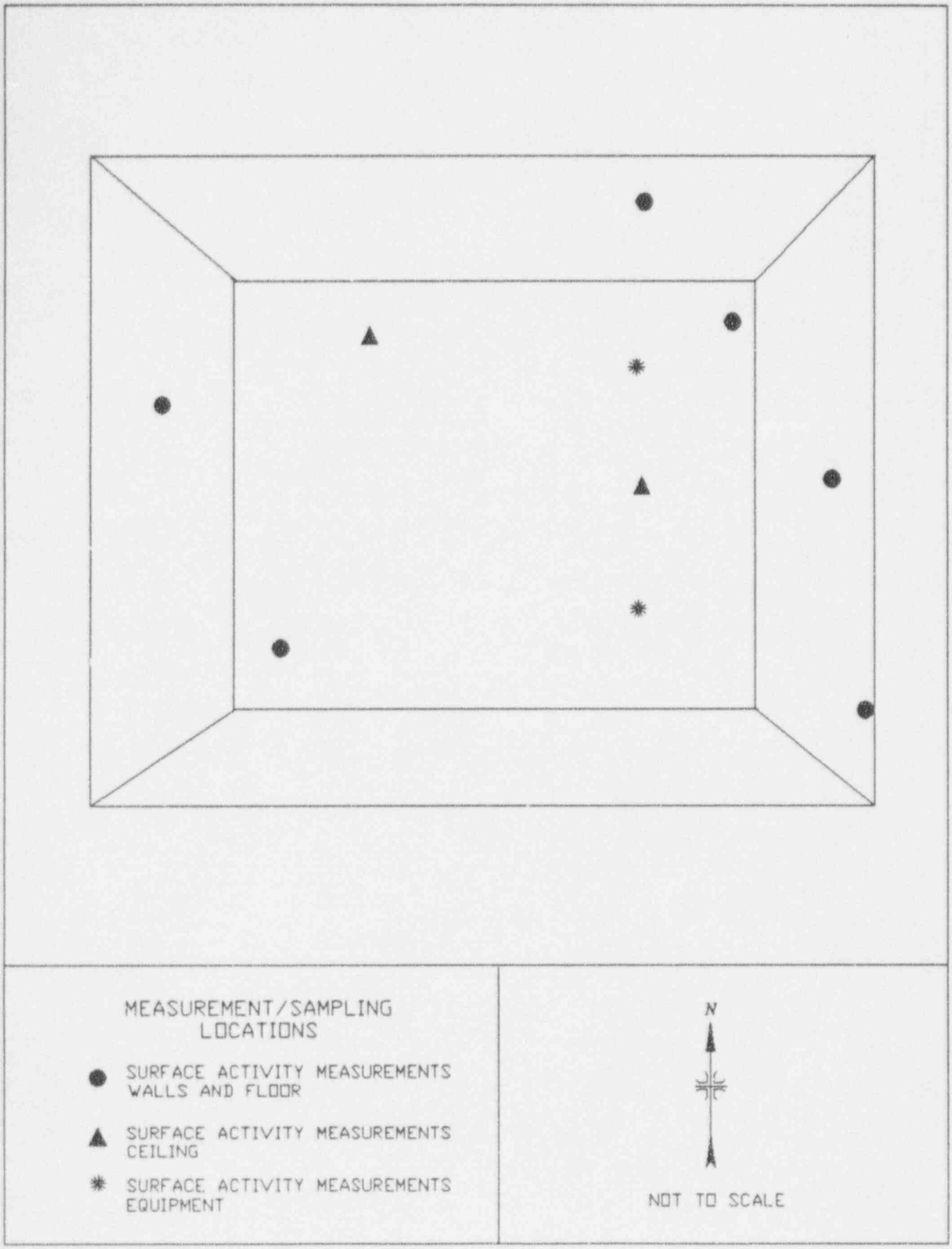


FIGURE 23: Turbine Building, Influent Drain System, Low Conductivity and Salt Water Drains, Tank 186B (SU054X03) - Measurement and Sampling Locations

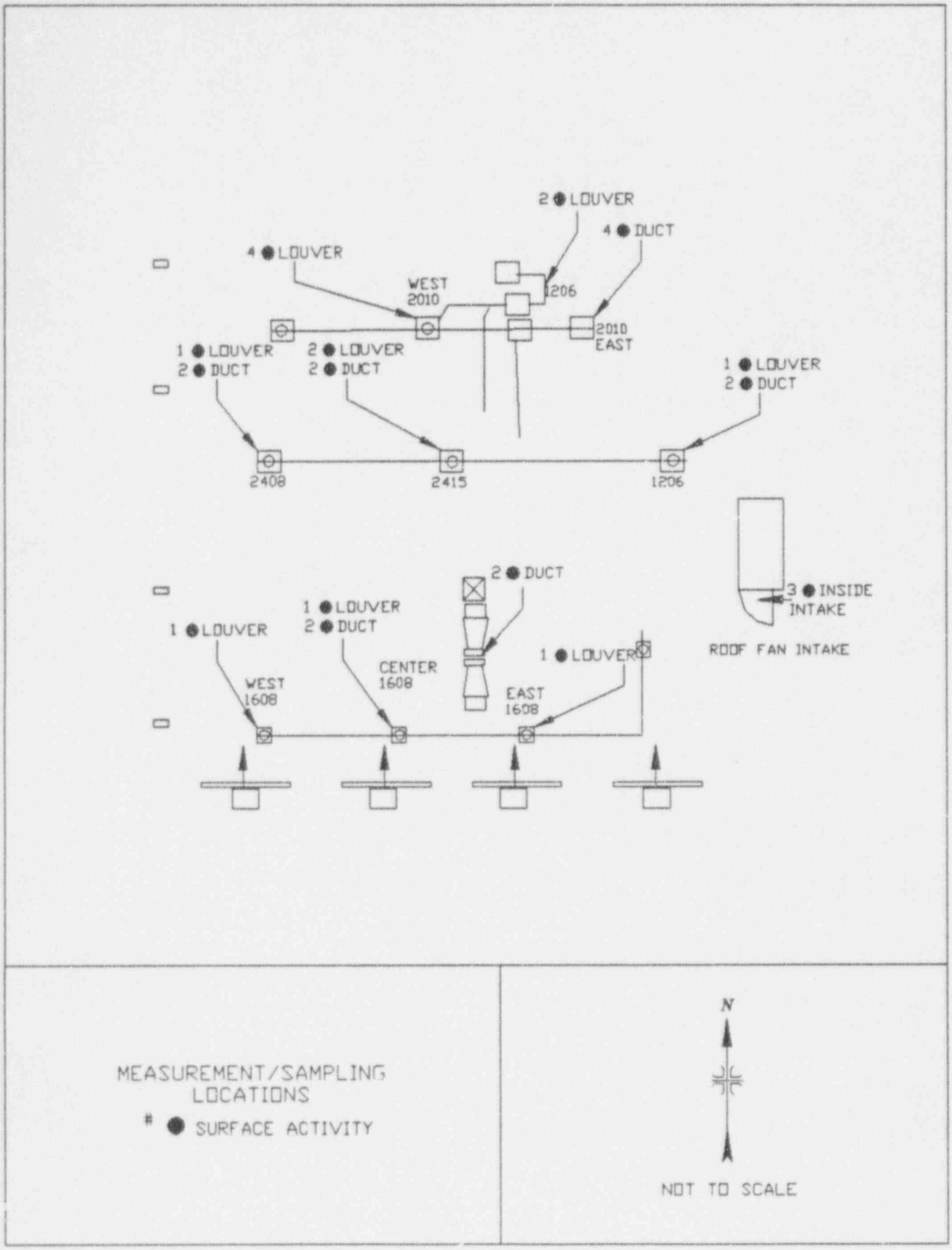


FIGURE 24: Secondary Access Facility, Ventilation System (SU071) - Measurement and Sampling Locations

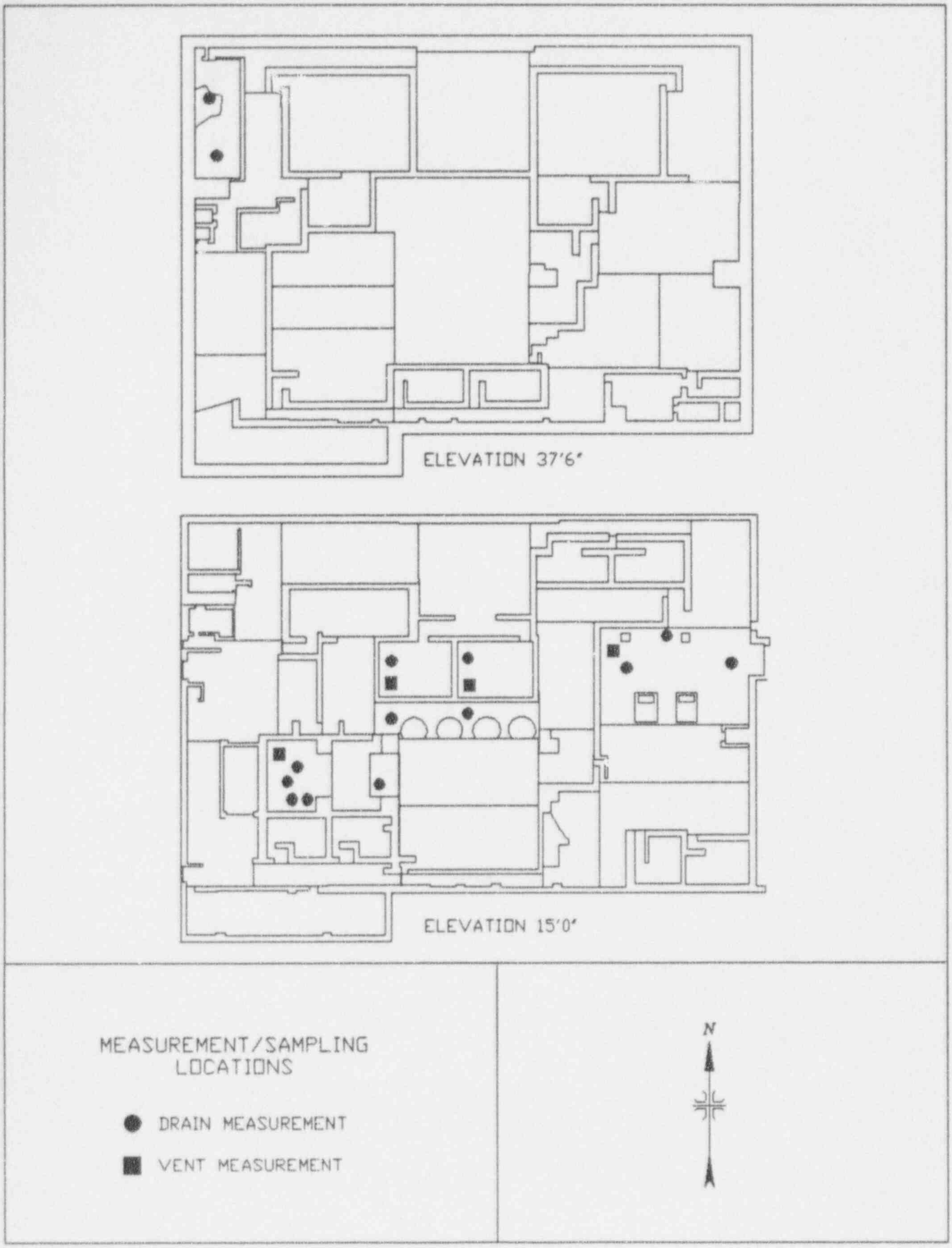


FIGURE 25: Turbine Building, 15' and 37'6" Elevations - Drains (SU014) and Vents - Measurement and Sampling Locations

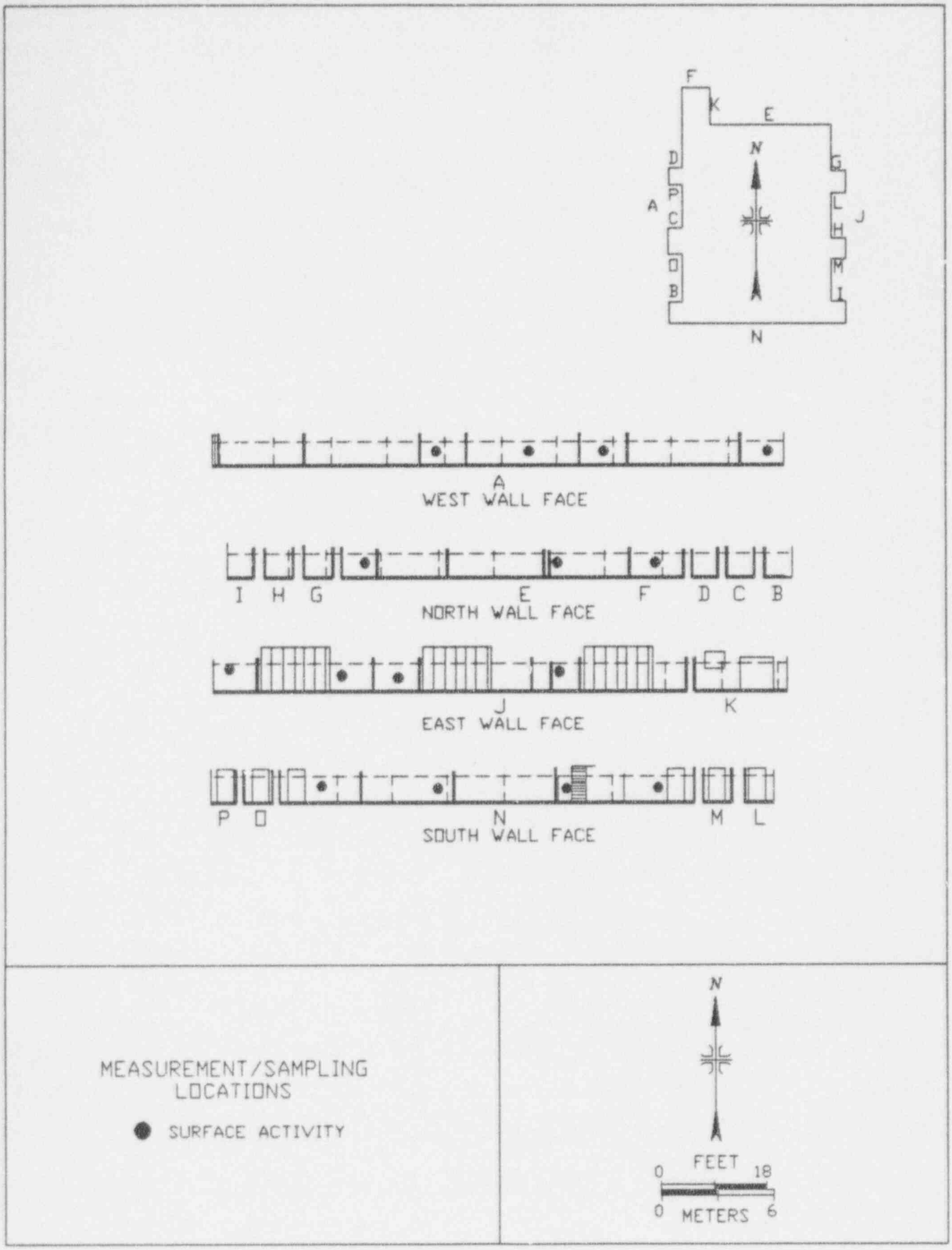


FIGURE 26: Secured Area, Colt Emergency Diesel Generator Building Exterior (SE002) - Measurement and Sampling Locations

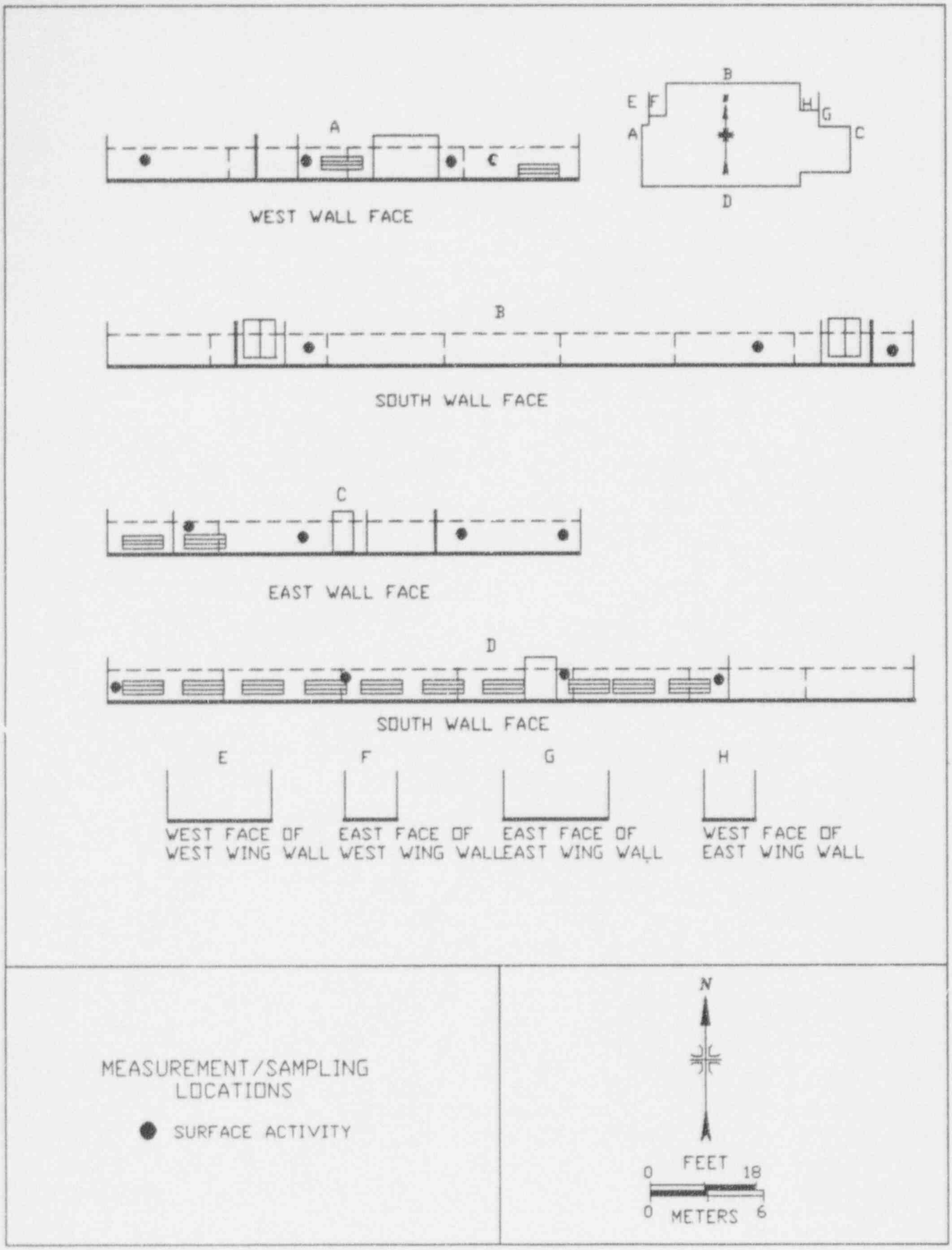


FIGURE 27: Secured Area, Chlorine Pumphouse Exterior (SE002) - Measurement and Sampling Locations

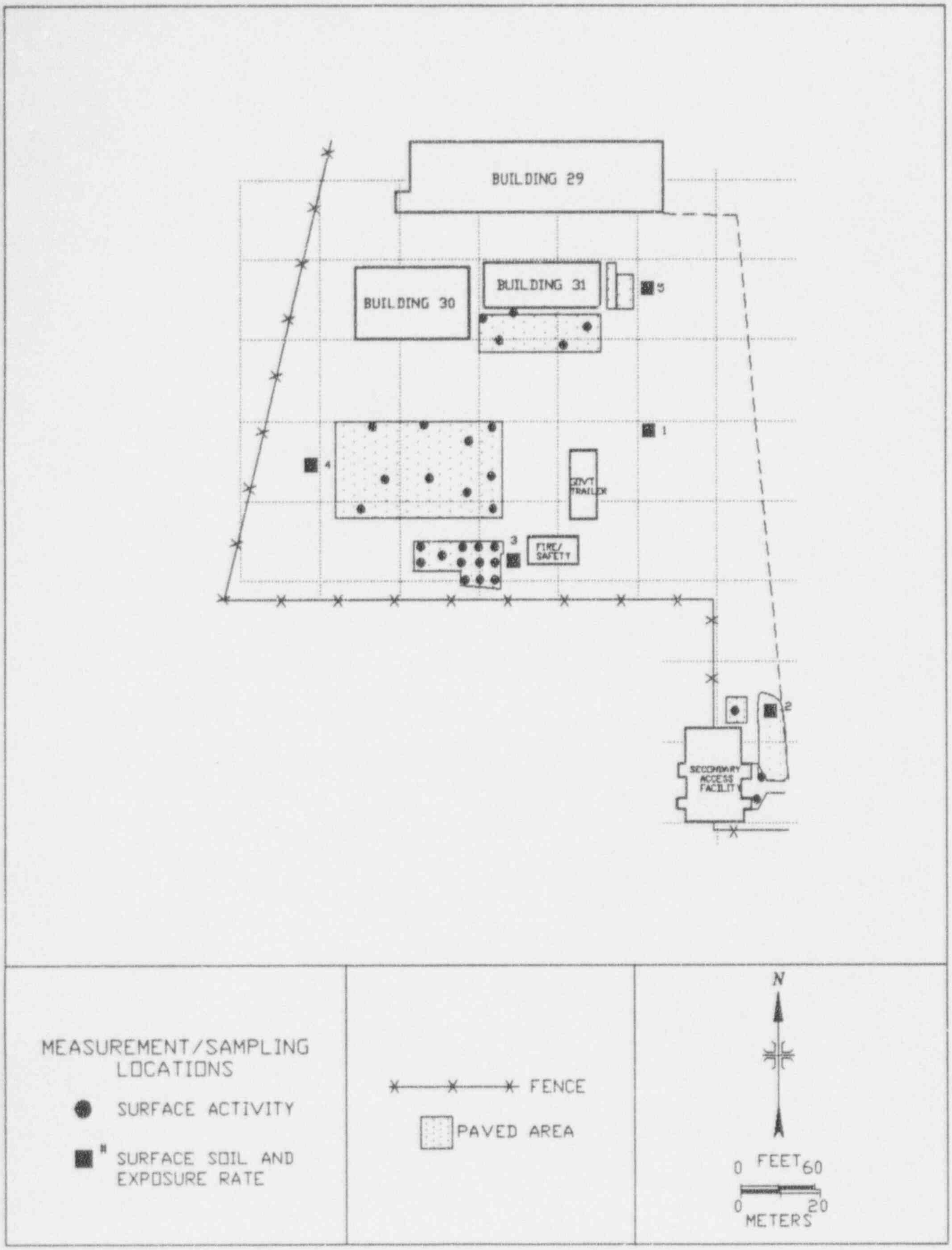


FIGURE 28: Secured Area West (SG003) - Measurement and Sampling Locations

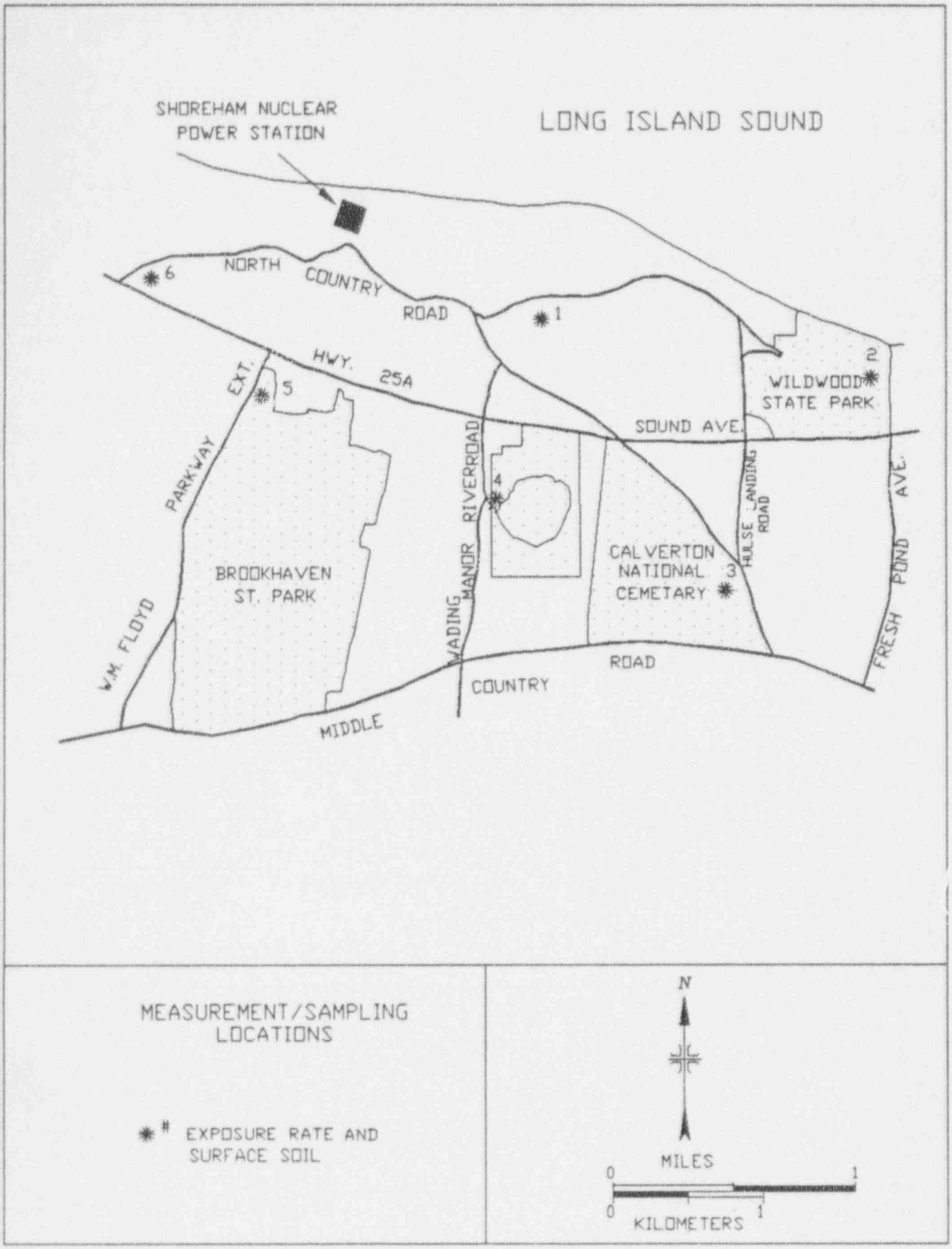


FIGURE 29: Background Soil Sampling and Exposure Rate Measurement Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS
TURBINE BUILDING, SITE GROUNDS, AND SITE EXTERIORS
SHOREHAM NUCLEAR POWER STATION
BROOKHAVEN, NEW YORK

Location[a]	Number of Measurement Locations	Total Activity Range ($dpm/100\ cm^2$) Beta[b]	Removable Activity Range ($dpm/100\ cm^2$)	
			Alpha[c]	Beta[d]
TB016 North Condenser Hallway	30	-570 to 440	-1 to 4	-5 to 4
TB017 West Condenser Bay	30	-680 to 290	-1 to 6	-6 to 5
TB031 Steam Seal Evaporator Room	30	-290 to 860	-1 to 4	-4 to 7
TB035 Turbine Bldg. Truck Bay	30	-690 to 440	-1 to 4	-5 to 7
TB060 Chemistry Laboratory	30	-430 to 770	-1 to 4	-5 to 6
TB081 Re-heater Area-East	30	-800 to 2,100	-1 to 4	-7 to 16
TB082 Re-heater Area-West	30	-540 to 470	-1 to 4	-4 to 7
TB089 Black Battery Charger Room	30	-140 to 1400	-1 to 4	-6 to 7
SU005 Feedwater Control	5	-680 to 80	-1 to 1	-5 to 3
SU014 Influent Drain Piping	16	-730 to 040	-1 to 4	-5 to 5
SU014X03 Tank 12	6	-650 to -340	-1 to 4	-7 to 7
SU024 Main Steam Components	25	-720 to 320	-1 to 4	-7 to 7
SU025X02 Main Condenser TB-15	30	-910 to 5,100	-1 to 4	-5 to 8
SU032 Lube Oil Sump Tank 91	3	-570 to 230	-1 to 1	-4 to 4
SUO34 Valve 035C	3	-490 to -460	-1 to 1	-1 to 0

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS
TURBINE BUILDING, SITE GROUNDS, AND SITE EXTERIORS
SHOREHAM NUCLEAR POWER STATION
BROOKHAVEN, NEW YORK

Location[a]	Number of Measurement Locations	Total Activity Range (dpm/100 cm^2) Beta[b]	Removable Activity Range (dpm/100 cm^2)	
			Alpha[c]	Beta[d]
SU046 Condensate Trans and Storage	30	-390 to 5,800	-1 to 4	-7 to 16
SU054X03 Tank 186	10	-610 to 470	-1 to 4	-5 to 4
SU071X70 Secondary Access Vents	30	-600 to 270	-1 to 4	-4 to 13
SE002 General Bldg. "A" Exterior	30	-760 to 270	-1 to 10	-4 to 7
SG003 Secured Area West, Paved Area	30	76 to 1,300	-1 to 6	-7 to 7
Miscellaneous Turbine Building Vents	4	-400 to 240	-1 to 1	-4 to 3

[a]Refer to Figures 7 through 14 and 16 through 28.
[b]MDAs = 990 to 1300 dpm/100 cm^2
[c]MDA = 12 dpm/100 cm^2
[d]MDA = 16 dpm/100 cm^2

TABLE 2

INTERIOR EXPOSURE RATES
TURBINE BUILDING
SHOREHAM NUCLEAR POWER STATION
BROOKHAVEN, NEW YORK

Location[a]	Number of Measurement Locations	Exposure Rate Range at 1 m (μR/h)
Background		
Colt Building	10	4 to 5
Turbine Building		
TB016 North Condenser Hallway	9	3 to 4
TB017 West Condenser Bay	6	4 to 5
TB031 Steam Seal Evaporator Room	15	4 to 5
TB035 Truck Bay	24	4 to 6
TB060 Chemistry Laboratory	13	5 to 6
TB081 Re-heater Area-East	12	5 to 6
TB082 Re-heater Area-West	10	4 to 5
TB089 Black Battery Charger Room	12	6 to 7

[a]Refer to Figures 7 through 15.

TABLE 3

EXTERIOR EXPOSURE RATES
AND
Co-60 CONCENTRATIONS IN SURFACE SOIL SAMPLES
SHOREHAM NUCLEAR POWER STATION
BROOKHAVEN, NEW YORK

Location[a]	Exposure Rate at 1 m (μR/h)	Co-60 Concentration (pCi/g)
Background		
Location #1	6	<0.1
Location #2	7	<0.1
Location #3	9	<0.1
Location #4	7	<0.1
Location #5	8	<0.1
Location #6	9	<0.1
Secured Area West (SG003)		
Location #1	9	<0.1
Location #2	8	<0.1
Location #3	8	0.1 ± 0.1[b]
Location #4	9	<0.1
Location #5	8	<0.1

[a]Refer to Figure 28 and 29.
[b]Uncertainties represent the 95% confidence level, based only on counting statistics.

TABLE 4

CONFIRMATORY RADIOLOGICAL STATUS SUMMARY—STRUCTURES, SITE GROUNDS, AND SITE EXTERIORS SHOREHAM NUCLEAR POWER STATION BROOKHAVEN, NY

Radiological Summary	Survey Unit[a]				
	TB016	TB017	TB031	TB035	TB060
Total Beta Activity (dpm/100 cm^2)					
# of Direct Measurements	30	30	30	30	30
Mean ($\bar{X}$)	-62	-130	7	-110	200
LIPA $\bar{X}$	-63	-480	-16	24	190
μ_α	30	-52	77	-27	300
Conditions and 5,000/15,0000 dpm/100 cm^2 Guidelines Satisfied	Yes	Yes	Yes	Yes	Yes
Removable Beta Activity (dpm/100 cm^2)[b]					
# of Smears	30	30	30	30	30
Mean ($\bar{X}$)	-1.2	0.1	0.1	-0.4	0.1
LIPA $\bar{X}$	1.9	1.6	2.9	3.7	1.7
μ_α	-0.5	1.0	1.1	0.6	1.0
Conditions and 1,000 dpm/100 cm^2 Guideline Satisfied	Yes	Yes	Yes	Yes	Yes
Exposure Rates at 1 m (μR/h)					
# of Exposure Rate Measurements	9	6	15	23	13
Net Mean ($\bar{X}$)	-0.7	-0.5	-0.1	0.4	0.9
LIPA $\bar{X}$	-0.1	0.2	0.2	0.1	0.7
μ_α	-0.5	-0.2	0.2	0.6	1.1
Conditions and 5 μR/h Above Background Guideline Satisfied	Yes	Yes	Yes	Yes	Yes

TABLE 4 (Continued)

CONFIRMATORY RADIOLOGICAL STATUS SUMMARY—STRUCTURES, SITE GROUNDS, AND SITE EXTERIORS SHOREHAM NUCLEAR POWER STATION BROOKHAVEN, NY

Radiological Summary	Survey Unit[a]				
	TB081	TB082	TB089	SE002	SG003
Total Beta Activity (dpm/100 cm²)					
# of Direct Measurements	30	30	30	30	30
Mean ($\bar{X}$)	-290	-84	370	-270	4
LIPA $\bar{X}$	450	65	-280	430	440
μ_α	-110	-2	480	-180	87
Conditions and 5,000/15,0000 dpm/100 cm² Guidelines Satisfied	Yes	Yes	Yes	Yes	Yes
Removable Beta Activity (dpm/100 cm²)[b]					
# of Smears	30	30	30	30	30
Mean ($\bar{X}$)	0.6	-0.2	-1.4	0.8	-1.0
LIPA $\bar{X}$	3.5	4.1	0.7	2.1	2.6
μ_α	1.9	0.6	- 0.4	1.8	-0.1
Conditions and 1,000 dpm/100 cm² Guideline Satisfied	Yes	Yes	Yes	Yes	Yes
Exposure Rates at 1 m (μR/h)					
# of Exposure Rate Measurements	12	10	12	---[c]	---
Net Mean ($\bar{X}$)	0.5	-0.3	2.3	---	---
LIPA $\bar{X}$	-0.1	-0.5	1.0	---	---
μ_α	0.7	-0.1	2.4	---	---
Conditions and 5 μR/h Above Background Guideline Satisfied	Yes	Yes	Yes	---	---

[a]Refer to Figures 7 through 14 and 24 through 27.
[b]All alpha removable activity was less than 12 dpm/100 cm².
[c]--- = Measurements not performed.

TABLE 5

CONFIRMATORY RADIOLOGICAL STATUS SUMMARY—SYSTEMS
TURBINE BUILDING
SHOREHAM NUCLEAR POWER STATION
BROOKHAVEN, NY

Radiological Summary	Survey Unit[a]				
	SU005	SU014-Drains	SU014X03	SU024	SU025X2
Total Beta Activity (dpm/100 cm²)					
# of Direct Measurements	5	16	6	25	30
Mean ($\bar{X}$)	-340	-340	-490	-240	52
LIPA $\bar{X}$	380	-NA[c]	-210	350	-120
μ_α	-57	-230	-370	-170	480
Conditions and 5,000/15,000 dpm/100 cm² Guidelines Satisfied	Yes	Yes	Yes	Yes	Yes
Removable Beta Activity (dpm/100 cm²)[b]					
# of Smears	5	16	6	25	30
Mean ($\bar{X}$)	- 0.1	-1.1	-1.0	-0.8	0.3
LIPA $\bar{X}$	2.1	NA	6.2	6.2	7.2
μ_α	3.1	0.1	0.3	0.2	1.4
Conditions and 1,000 dpm/100 cm² Guideline Satisfied	Yes	Yes	Yes	Yes	Yes

TABLE 5 (Continued)

CONFIRMATORY RADIOLOGICAL STATUS SUMMARY—SYSTEMS
TURBINE BUILDING
SHOREHAM NUCLEAR POWER STATION
BROOKHAVEN, NY

Radiological Summary	Survey Unit[a]				
	SU032	SU034	SU046	SU054X3	SU071X70
Total Beta Activity (dpm/100 cm^2)					
# of Direct Measurements	3	3	34	10	30
Mean ($\bar{X}$)	-410	270	300	-220	-90
LIPA $\bar{X}$	290	300	560	-380	15
μ_α	-120	-390	630	7	-29
5,000/15,000 dpm/100 cm^2 Guidelines Satisfied	Yes	Yes	Yes	Yes	Yes
Removable Beta Activity (dpm/100 cm^2)					
# of Smears	3	3	30	10	30
Mean ($\bar{X}$)	-0.7	-0.5	1.2	-0.4	0.3
LIPA $\bar{X}$	2.9	1.9	13.5	4.9	1.0
μ_α	1.5	0.7	2.7	1.3	2.2
Conditions/1,000 dpm/100 cm^2 Guideline Satisfied	Yes	Yes	Yes	Yes	Yes

[a]Refer to Figure 16 through 25.
[b]All alpha removable activity was less than 12 dpm/100 cm^2.
[c]NA = not applicable

REFERENCES

1. Long Island Lighting Company, "Shoreham Nuclear Power Station Site Characterization Program Final Report," May 1990.

2. T. J. Vitkus, ORISE, "Confirmatory Survey of the Turbine Internal Components, Shoreham Nuclear Power Station, Brookhaven, New York," July 1993.

3. J. D. Berger, Oak Ridge Associated Universities, Draft - "Manual for Conducting Radiological Surveys in Support of License Termination," NUREG/CR-5849, June 1992.

4. Long Island Power Authority, "Shoreham Decommissioning Project, Termination Survey Plan, Revision 1," April, 1993.

5. Long Island Power Authority, "Shoreham Decommissioning Project Termination Survey Final Report, Volumes 1 through 5," September, 1993.

6. Letter from T. J. Vitkus, ORISE to D. Fauver, U.S. Nuclear Regulatory Commission, "Final Confirmatory Survey Plan for the Shoreham Nuclear Power Station, Brookhaven, New York - Docket File No. 50-322," November 4, 1993.

7. Letter from D. N. Fauver, U.S. Nuclear Regulatory Commission, to T. Vitkus, ORISE, July 1, 1993.

8. Letter from M. R. Landis, ORISE to D. Fauver, U.S. Nuclear Regulatory Commission, "Shoreham Decommissioning Project, Termination Survey Plan, Revision O, Shoreham Nuclear Power Station, October 1992," January 12, 1993.

9. Long Island Power Authority, "Shoreham Decommissioning Project Termination Survey Final Report," Volume 1, June 1994.

10. U.S. Nuclear Regulatory Commission, "Guidance and Discussion of Requirements for an Application to Terminate a Non-Power Reactor Facility Operating License," Revision 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 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)

Ludlum Gas Proportional Detector
Model 43-68
Effective Area, 100 cm^2
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Reuter-Stokes Pressurized Ion 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)

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 (15.5 cm^2, 59 cm^2 or 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	—	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 beta activity levels were primarily performed using GM detectors with portable ratemeter-scalers.

Count rates (cpm), which were integrated over 1 minute in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the net rate by the 4 π efficiency and correcting for the active area of the detector. The beta activity background count rates for the GM detectors ranged from 22 to 42 cpm. Beta efficiency factors ranged from 0.16 to 0.18 for the GM detectors. The effective window for the GM detectors was 15.5 cm^2.

Surface activity measurements which exceeded the normal background distribution were corrected for the Fe-55 contribution by multiplying the dpm/100 cm^2 field activity level by a factor of 1.2. The instrument response level at which the detector output could be considered above background was defined as the critical level (L_c). This level was defined for each detector/instrument combination as follows:

$$L_c = \frac{1.96\sqrt{\frac{Sample\ count\ rate}{Sample\ count\ time} \cdot \frac{Background\ count\ rate}{Background\ count\ time}}}{(Detector\ Efficiency)\ (Detector\ Geometry)}$$

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

Removable Activity

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

Gamma Spectrometry

Soil and sludge samples 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 and ranged from 556 to 1238 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. The energy peak used for determining the activity of the radionuclide of concern was:

Co-60 1.173 MeV

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 $[2.71 + (4.66\sqrt{BKG})]$. 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 radionuclides 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 of the Environmental Survey and Site Assessment Program:

- Survey Procedures Manual, Revision 7 (May 1992)
- Laboratory Procedures Manual, Revision 8 (August 1993)
- Quality Assurance Manual, Revision 6 (July 1993)

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, traceable to NIST, when such standard were available. In cases where they were not available, standards of an industry recognized organization were 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 DOE/EML 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

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 [illegible] the technical specifications to assure that prop[illegible] 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 organization for

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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 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 pertinent levels should be reduced proportionally and the entire surface should be wiped.