DRAFT REPORT

CONFIRMATORY SURVEY FOR THE FORT ST. VRAIN NUCLEAR STATION PUBLIC SERVICE COMPANY OF COLORADO PLATTEVILLE, COLORADO

[DOCKET NO. 50-267]

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

ORISE
OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION
Environmental Survey and Site Assessment Program
Environmental and Health Sciences Division

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FOR THE
FORT ST. VRAIN NUCLEAR STATION
PUBLIC SERVICE COMPANY OF COLORADO
PLATTEVILLE, COLORADO

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JANUARY 1997

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.

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FOR THE
FORT ST. VRAIN NUCLEAR STATION
PUBLIC SERVICE COMPANY OF COLORADO
PLATTEVILLE, COLORADO

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ACKNOWLEDGMENTS

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ABBREVIATIONS AND ACRONYMS

μR/h	microroentgens per hour
ASME	American Society of Mechanical Engineers
cm	centimeter
cm^2	square centimeter
cpm	counts per minute
DOE	Department of Energy
dpm/100 cm^2	disintegrations per minute/100 square centimeters
EML	Environmental Measurements Laboratory
EPA	Environmental Protection Agency
ESSAP	Environmental Survey and Site Assessment Program
FSV	Fort St. Vrain
GM	Geiger-Muller
HTDN	hard-to-detect nuclide
HTGR	High Temperature Gas-Cooled Reactor
m	meter
m^2	square meter
mm	millimeter
MDC	minimum detectable concentration
MeV	million electron volts
MWe	Megawatts electric
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
PCRV	prestressed concrete reactor vessel
PIC	pressurized ionization chamber
PSC	Public Service Company of Colorado

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INTRODUCTION

The Public Service Company of Colorado (PSC) operated a 330 MWe High Temperature Gas Cooled Reactor (HTGR) from July 1979 until August 1989. The plant, designated as the Fort St. Vrain Nuclear Station (FSV), was authorized for construction on September 17, 1968 when the U.S. Nuclear Regulatory Commission (NRC) issued a provisional construction permit. Construction was completed in December 1973 and a facility operating license, License No. DPR-34, Docket No. 50-267, was granted on December 21, 1973. Initial fuel loading commenced on December 26, 1973 and initial criticality was achieved January 31, 1974. After a prolonged period of startup testing, low-power operation and plant modifications, the plant was committed for commercial operation on July 1, 1979. Full power was achieved November 6, 1981 (PSC 1995a).

In the nuclear steam supply system for FSV, heat was produced by fission in the HTGR utilizing a uranium-thorium fuel cycle. Graphite was used for the moderator, core structure, and reflector. High temperature helium was used as the primary coolant to produce superheated and reheated steam at a temperature of 1,000°F to match conventional thermal station conditions. The entire nuclear steam supply system, including the reactor core, graphite moderator and reflector, steam generators and helium circulators, was contained within a Prestressed Concrete Reactor Vessel (PCRV).

During the operational period, FSV operated for approximately 890 effective full-power days; FSV was shut down on August 18, 1989. The PSC Board of Directors reviewed and confirmed the Executive Management decision that FSV would not be restarted, and that PSC would pursue decommissioning of FSV. The decision to permanently shut down and decommission FSV was based on related technical and financial considerations. Problems were identified with the control

rod drive assemblies and the steam generator steam ring headers that presented significant technical obstacles which could be overcome, but at a significant financial cost and time commitment. In addition, due to the uniqueness of the HTGR fuel cycle, the cost to purchase new fuel was prohibitive. This, in conjunction with low plant availability and correspondingly high operating costs, made continued operation of FSV impractical.

PSC's objective is the dismantlement and decommissioning of FSV to release all site areas for unrestricted use. To accomplish this, a portion of the PCRV structure and the radioactive balance-of-plant equipment that exceed the limits for unrestricted use was decontaminated or removed as described in the Fort St. Vrain Decommissioning Plan. In May 1991, the NRC granted a 10 CFR 50 Possession Only License. On November 23, 1992, the NRC issued the Order to Authorize Decommissioning of Fort St. Vrain and Amendment No. 85 to Possession Only License No. DPR-34 (PSC 1995b).

The FSV facility was largely left intact following decommissioning; dismantlement of structures was confined to the PCRV, and portions of the Reactor Building, Turbine Building, and Liquid Waste System.

Following defueling, the PCRV contained the majority of the remaining radioactive material inventory. Portions of the PCRV concrete were activated due to direct irradiation from the reactor core, and concrete exceeding unrestricted use limits was removed prior to final survey and disposed of as radioactive waste at a licensed radioactive waste disposal facility. Thus, the radioactive source term at FSV was primarily a result of neutron activation of both metallic and concrete components of the PCRV and neutron activation of impurities contained in graphite components of the PCRV. These activation products included beta-gamma emitters such as Co-60, Eu-152, and Eu-154, and low-energy beta and x-ray emitters such as H-3, C-14, and Fe-55. It should be noted that H-3 and Fe-55 were the largest contributors to the total radionuclide inventory (PSC 1995a).

FSV's final survey has included all pertinent structures, surfaces, systems and components, concentrating on those previously identified as contaminated or potentially contaminated during the dismantlement/decommissioning phases. The FSV final status survey included:

- Sampling of soil, pavement, water, and liquid effluent ditch and pond sediment for radionuclide analysis and measurement of gamma exposure rate outside the restricted area of PSC property,

- Sampling of soil, basin sediment, pavement and water for radionuclide analysis and measurement of gamma exposure rate inside the restricted area of PSC property,

- Radiological surveys of the PCRV and Reactor Building, and

- Radiological surveys of the Turbine Building, Radwaste Compactor Building, New Fuel Storage Building, Radiochemistry Laboratory, Helium Transfer and Storage System, and Liquid Radwaste System.

At the request of the NRC's Division of Waste Management, the Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) performed an independent confirmatory survey of the repower area in March 1995 at the Fort St. Vrain site in Platteville, Colorado (ORISE 1995a). Subsequent independent survey activities at FSV included licensee survey package reviews, confirmatory surface scans, and comparison surface activity measurements (e.g., side-by-side measurements) performed from September 25 through 27, 1995 (ORISE 1996a). During the period January 22 through 25, 1996, ESSAP performed instrument comparison activities—including side-by-side surface activity measurements and surface scans—and reviews of the licensee's embedded piping program and use of *in situ* gamma spectrometry for determining the licensed material contribution to exposure rate (ORISE 1996b). Most recently, ESSAP performed independent confirmatory surveys during the period September 30 through October 3, 1996. Specifically, ESSAP performed surface scans, direct measurements of surface activity, and exposure rate measurements; and reviewed the licensee's

hard-to-detect nuclide (HTDN) assessment program. A preliminary report describing these activities was prepared by ESSAP and submitted to the NRC in a letter dated October 22, 1996 (ORISE 1996c).

SITE DESCRIPTION

The FSV facility is located approximately 56 kilometers (35 miles) north of Denver and 5.6 kilometers northwest of the town of Platteville, in Weld County, Colorado (Figure 1). The site is located in an agricultural area with gently rolling hills. Grade elevation at the plant is 1,460 meters (4,790 feet) above sea level. The site consists of 6995 hectares (17,300 acres) owned by PSC, identified as the Owner-Controlled Area, of which approximately one square mile was designated as the exclusion area during plant operation. Farming has been continued on Owner-Controlled areas of the site, but there are no farming operations or permanent residences located within the Restricted Area. The Restricted Area is surrounded by a security fence, and access is controlled for purposes of protection of individuals from exposure to radiation.

The station is located approximately two miles south of the confluence of the South Platte River and the St. Vrain Creek. Neither of these two streams are considered navigable. Cooling for the plant is provided by mechanical draft cooling towers. Make-up to the cooling towers is obtained from the two streams, and is supplemented by shallow well water. Nineteen shallow wells are located on the site. The licensee also owns surface water rights in four irrigation ditches which traverse portions of the site.

The major structures within the Restricted Area include the Reactor Building which contains the PCRV, Turbine Building, Radwaste Compactor Building, New Fuel Storage Building, Technical Support Building which contains the Radiochemistry Laboratory, Mechanical Draft Cooling Towers, Warehouse and Construction Workshops, Evaporation Ponds, and the Electrical Switchyard (Figure 2). The ground surface covering within the Restricted Area is composed primarily of gravel and vegetation, with smaller portions devoted to concrete or asphalt roadways and laydown areas.

OBJECTIVES

The objectives of the confirmatory survey were to provide independent contractor field data reviews and radiological data for use by the NRC in evaluating the adequacy and accuracy of the licensee's procedures and final status survey results.

DOCUMENT REVIEW

ESSAP reviewed the licensee's final status survey documentation for those survey units contained within Volumes 1 through 5 (PSC 1996). Additional documentation related to the licensee's HTDN program were reviewed during the on-site visit. Subsequent to the confirmatory survey, ESSAP reviewed the licensee's final status survey documentation contained in Volumes 6 through 11, including any revisions made to Volumes 1 through 5. Documents were reviewed for adequacy, accuracy, completeness, and consistency. Data were reviewed for appropriateness of calculations and interpretations relative to the guidelines.

PROCEDURES

During the period September 30 through October 3, 1996, ESSAP performed confirmatory survey activities at the Fort St. Vrain site in Platteville, Colorado. Survey activities included technical review of the HTDN program and independent confirmatory survey activities. Eight survey units were selected by the NRC, with input from ESSAP, for confirmatory survey activities—including surface scans, surface activity measurements, and exposure rate measurements. The selected survey units were (convention used—survey group/survey unit): A0007/BZ003, B0006/FZ001, B0012/FZ002, C0004/WZ001, C0009/BZ001, C0030/FZ001, C0031/SZ001, and D4800. In addition, surface scans were performed in NRC-selected survey units F0015, F0039, F0077, F0084 (and a portion of the adjacent F0115), F0126, and the refuel floor (Level 11 of the Reactor Building). The survey was in accordance with a plan dated September 16, 1996 (ORISE 1996d) submitted to and approved by the NRC's Division of Waste Management. Survey procedures were performed in accordance with the ORISE/ESSAP Survey Procedures and Quality Assurance Manuals (ORISE 1995b and c). This report summarizes the procedures and results of the survey.

Survey Procedures

The following procedures apply to survey units selected for independent confirmatory activities.

Reference System

The reference systems established by FSV were used by ESSAP for referencing measurement and sampling locations. Measurement and sampling locations on ungridded surfaces were referenced to prominent building features or the existing grid. All measurement and sampling locations were recorded on maps prepared by the licensee for each survey unit.

Surface Scans

Surface scans for beta and gamma activity were performed over 50 to 100% of floor and lower wall surfaces and up to 50% of equipment surfaces within each of the eight building surface and structure survey units selected for confirmatory survey activities. In addition, surface scans for alpha and beta activity were performed over accessible floor, lower wall, and equipment surfaces in NRC-selected survey units from Group F. Scans were performed using gas proportional, GM, and/or NaI scintillation detectors coupled to ratemeters or ratemeter-scalers with audible indicators. Locations of elevated direct radiation identified by scans were marked for further investigation.

Surface Activity Measurements

Construction material specific backgrounds for red brick, brick, carpet, concrete, metal, painted concrete, and tile brick were determined in areas of similar construction but without a history of radioactive material use. ESSAP used the licensee's plastic shield to obtain three local area background measurements in each survey unit. The total background level at each direct measurement location was obtained by adding the construction material background and the local area background.

For each building surface and plant system survey unit selected, ESSAP performed 10 to 30 direct measurements for total beta surface activity—resulting in a total of 171 direct measurements for surface activity. Additional direct measurements were performed at two locations of elevated direct radiation detected by surface scans. ESSAP also performed 20 direct measurements on miscellaneous concrete, metal, and wood debris from the demolition of the New Fuel Storage Building and Building 28, staged on the east side of the Fort St. Vrain site. Direct measurements were performed using GM or gas proportional detectors coupled to ratemeter-scalers.

A smear sample for determining the removable activity level was collected from each direct measurement location. Measurement and sampling locations are shown on Figures 3 through 15.

Exposure Rate Measurements

A total of 28 exposure rate measurements were performed within survey groups A0007, B0006, B0012, C0004, C0009, and C0030 (Figures 3 through 11). All exposure rates were measured at 1 meter above surfaces using a pressurized ionization chamber (PIC).

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ORISE's ESSAP laboratory in Oak Ridge, Tennessee for analysis and interpretation. Sample analyses were performed in accordance with the ORISE/ESSAP Laboratory Procedures Manual (ORISE 1995d). Smears were analyzed for gross alpha and gross beta activity using a low background gas proportional counter, and the results were converted to units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2).

Direct measurements for surface activity were converted to units of dpm/100 cm^2 and the surface activity results for each survey unit were statistically evaluated. The goal of the test was to determine, with a given confidence level, that the FSV surface activity levels were not biased low compared to ESSAP. The null hypothesis was stated that, in a survey unit, surface activities as calculated by FSV are greater than or equal to those determined by ESSAP, i.e., H_o: $\mu_{FSV} \geq \mu_{ESSAP}$. For surface activity comparisons, it should be noted that side-by-side comparisons performed during

a previous ESSAP site visit (ORISE 1996b) have determined that ESSAP's measurements are biased low as compared to FSV's measurements. Therefore, the statistical test may be considered as a measure for demonstrating that FSV's surface activity measurements are in statistical control, relative to the previously established relationship between ESSAP and FSV. This hypothesis was tested at the 95% confidence level (0.05 level of significance). If the hypothesis was rejected at that confidence lev..., the alternative hypothesis was to be accepted i.e., H_A: $\mu_{FSV} < \mu_{ESSAP}$. The test statistic, *t*, was calculated using the following equation:

$$t = \frac{\overline{X_E} - \overline{X_F}}{\sqrt{\frac{(n_E-1)S_E^2 + (n_F-1)S_F^2}{n_E + n_F - 2}\left(\frac{n_E + n_F}{n_E n_F}\right)}}$$

where:

$\overline{X_F}$ is the FSV mean for a survey unit

X_E is the ESSAP mean for the same survey unit

n_F is the number of FSV measurement/sampling data points

n_E is the number of ESSAP measurement/sampling data points

S_F, S_E are the standard deviations for the FSV and ESSAP measurement data, respectively.

The calculated *t* was compared to the critical value of Student's *t*-distribution (one-tailed) for the appropriate degrees of freedom at the 95% confidence level (0.05 level of significance). If the H_o: $\mu_{FSV} \geq \mu_{ESSAP}$ was rejected, then ESSAP conferred with the NRC as to the recommended approach.

Exposure rates were reported in microroentgens per hour (μR/h). Additional information regarding major instrumentation, sampling equipment, and analytical procedures is provided in Appendices A and B. The data generated were compared with the licensee's documentation and NRC guidelines established for release for unrestricted use (Appendix C).

FINDINGS AND RESULTS

DOCUMENT REVIEW

ESSAP reviewed the licensee's final status survey documentation and performed an on-site record review of the HTDN program. A preliminary report on the review of the licensee's HTDN program and a comment letter documenting the review of Volumes 1 through 5 were submitted to the NRC (ORISE 1996c and e). The licensee's documentation provides an adequate description of the radiological status of the surveyed areas and sufficient information on the HTDN program.

SURVEY RESULTS

Surface Scans

Surface scans performed within the eight survey units selected for confirmatory survey activities did not identify any locations of elevated direct radiation.

Surface scans performed within the Group F survey units identified two locations of elevated direct radiation. One location of elevated direct radiation was identified in survey unit F0077, an affected area located on Level 5 of the Reactor Building. The elevated direct radiation was located in a relatively inaccessible area behind an electrical raceway, and was identified with a GM detector.

The second location of elevated direct radiation was identified while performing scans in survey unit F0084, an affected area on Level 7 of the Reactor Building. The location of elevated direct radiation was identified on the exterior wall of the PCRV using a GM detector. It was subsequently determined by the licensee that this location of elevated direct radiation was actually within F0115, the survey unit immediately adjacent to, and above, survey unit F0084.

Surface Activity Levels

Surface activity levels for total beta activity and removable alpha and beta activity within the eight survey units selected for confirmatory survey activities are summarized in Table 1. Total beta activity in surveyed areas ranged from -890 to 1,200 dpm/100 cm^2. Removable activity was less than 9 dpm/100 cm^2 for gross alpha and ranged from <15 to 18 dpm/100 cm^2 for gross beta.

Surface activity levels for total beta activity and removable alpha and beta activity for the debris removed from Building 28 ranged from -370 to 1,400 dpm/100 cm^2. Removable activity was less than 9 dpm/100 cm^2 for gross alpha and less than 15 dpm/100 cm^2 for gross beta.

The surface activity level at the location of elevated direct radiation within F0077, as measured by the GM detector, was 19,600 dpm/100 cm^2. The surface activity level measured in survey unit F0115 with a 126-cm^2 gas proportional detector resulted in 9,210 dpm/100 cm^2. It should be noted that the licensee had also identified this elevated area with their scan survey and reported the surface activity level as 8,716 dpm/100 cm^2.

Exposure Rates

Exposure rates in surveyed areas are tabulated in Table 2 and ranged from 9 to 21 μR/h.

COMPARISON OF RESULTS WITH GUIDELINES

The primary contaminants of concern for this site are beta-gamma emitters resulting from the operation of the FSV facility. The applicable NRC guidelines for beta-gamma emitters in unaffected areas are (NRC 1974):

Total Activity

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

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

Removable Activity

1,000 dpm/100 cm^2

The NRC has approved site-specific allowable surface contamination guidelines for H-3 and Fe-55, particularly in activated concrete and steel (NRC 1994). These guidelines are:

Total Activity

200,000 dpm/100 cm^2, averaged over 1 m^2

600,000 dpm/100 cm^2, maximum in a 100 cm^2 area

Removable Activity

40,000 dpm/100 cm^2

FSV has assessed the radionuclide composition at the site and determined the site-specific surface activity guidelines for affected areas (PSC 1996). The effective beta-gamma surface contamination limits are:

Total activity

4,000 dpm/100 cm^2, averaged over a 1 m^2 area

12,000 dpm/100 cm^2, maximum in a 100 cm^2 area

Removable activity

750 dpm/100 cm^2

The two locations of elevated direct radiation identified by scans that exceeded the surface activity guideline were remediated by the licensee. No other direct measurements exceeded the average or maximum total surface activity guideline. All removable activity levels were below the appropriate guidelines.

A comparison of the ESSAP mean surface activity levels to the FSV mean surface activity levels using the Student's t-test is provided in Table 3. The results indicated that the ESSAP mean was statistically less than or equal to the respective mean determined by FSV in all 8 of the confirmatory survey units. That is, the null hypothesis (H_o: $\mu_{FSV} \geq \mu_{ESSAP}$) could not be rejected because the test statistic was less than the critical value of the t-test for each survey unit.

The exposure rates guideline, measured at 1 meter above the surface, is 5 μR/h above background (NRC 1988). Due to the variable exposure rate background, ESSAP elected to compare the total exposure rate (i.e., background was not subtracted) measured during the confirmatory survey to the licensee's total exposure rate. Comparison of total exposure rates from the six survey units evaluated showed general agreement between the licensee and ESSAP, with the exception of Survey Group B0012, survey unit FZ002 (Table 2).

SUMMARY

During the period September 30 through October 3, 1996, the Environmental Survey and Site Assessment Program of ORISE performed confirmatory survey activities at the Fort St. Vrain site in Platteville, Colorado. Survey activities included technical review of the hard-to-detect nuclide program and independent confirmatory survey activities, including surface scans, surface activity measurements, and exposure rate measurements.

The licensee's documentation provides an adequate description of the radiological status of the surveyed areas and sufficient information on the HTDN program. Confirmatory survey activities in eight survey units identified no locations of elevated direct radiation. Surface activity levels were below those reported by the licensee and therefore, demonstrated compliance with the surface activity guidelines. Statistical tests of data further support the conclusion that the licensee's surface activity levels were not biased low as compared to ESSAP's measurement results.

Comparison of scan results from the five survey units evaluated showed general agreement between the licensee and ESSAP. Overall, the confirmatory survey results for the surveyed areas are consistent with those of the licensee and support the conclusion that surface activity levels and exposure rates satisfy the guidelines for release for unrestricted use.

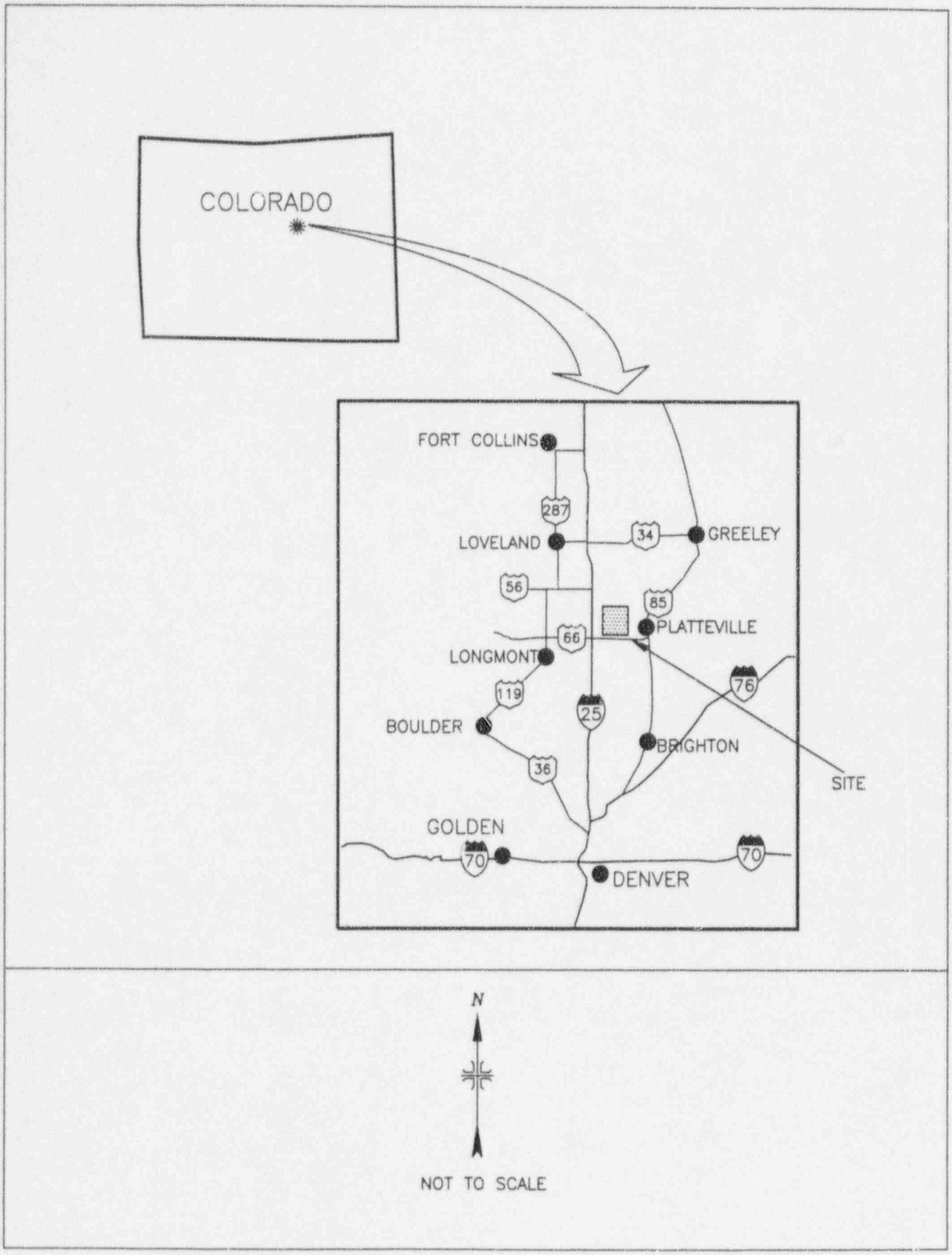

FIGURE 1: Location of the Fort St. Vrain Site – Platteville, Colorado

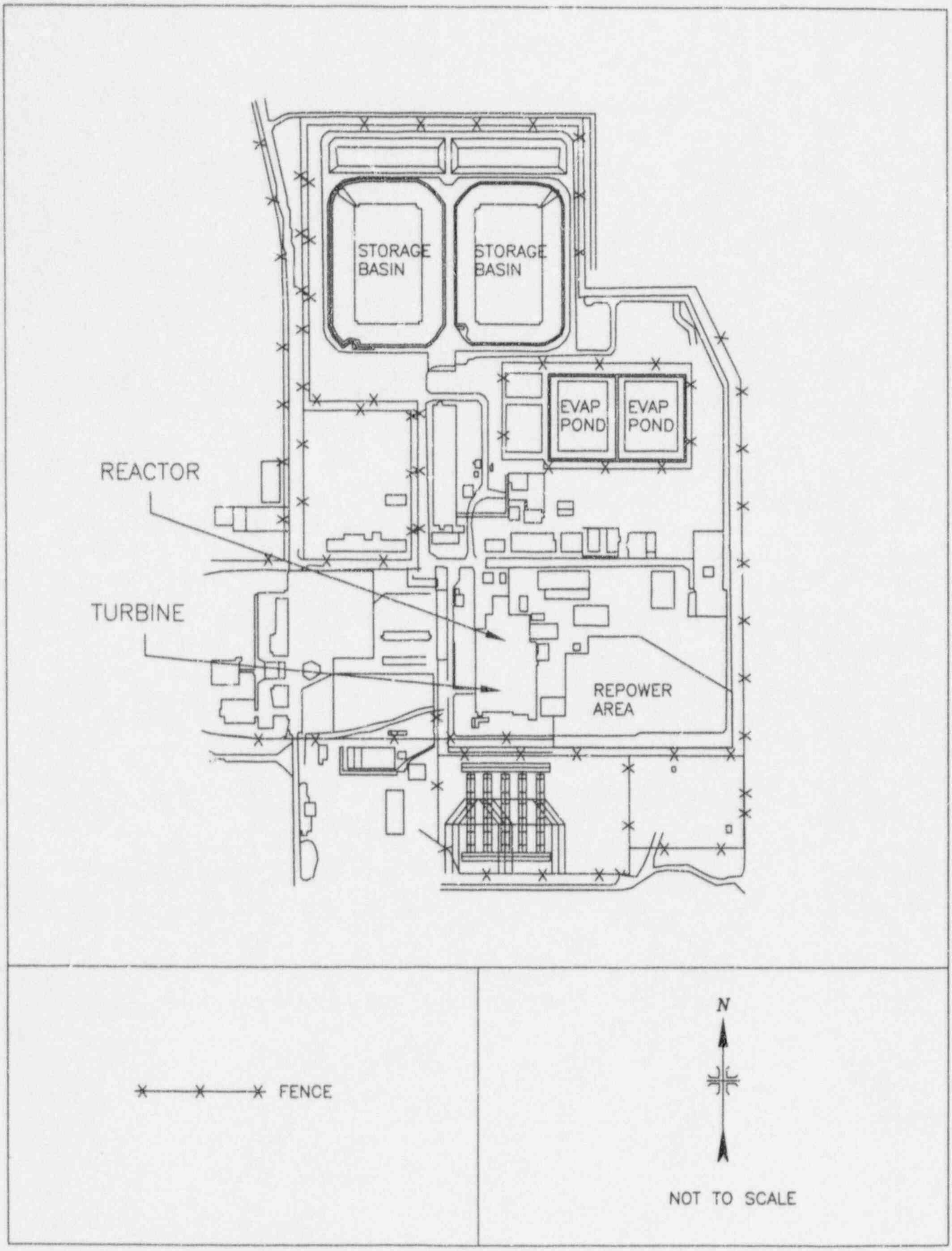

FIGURE 2: Plot Plan of the Fort S' Vrain Nuclear Station

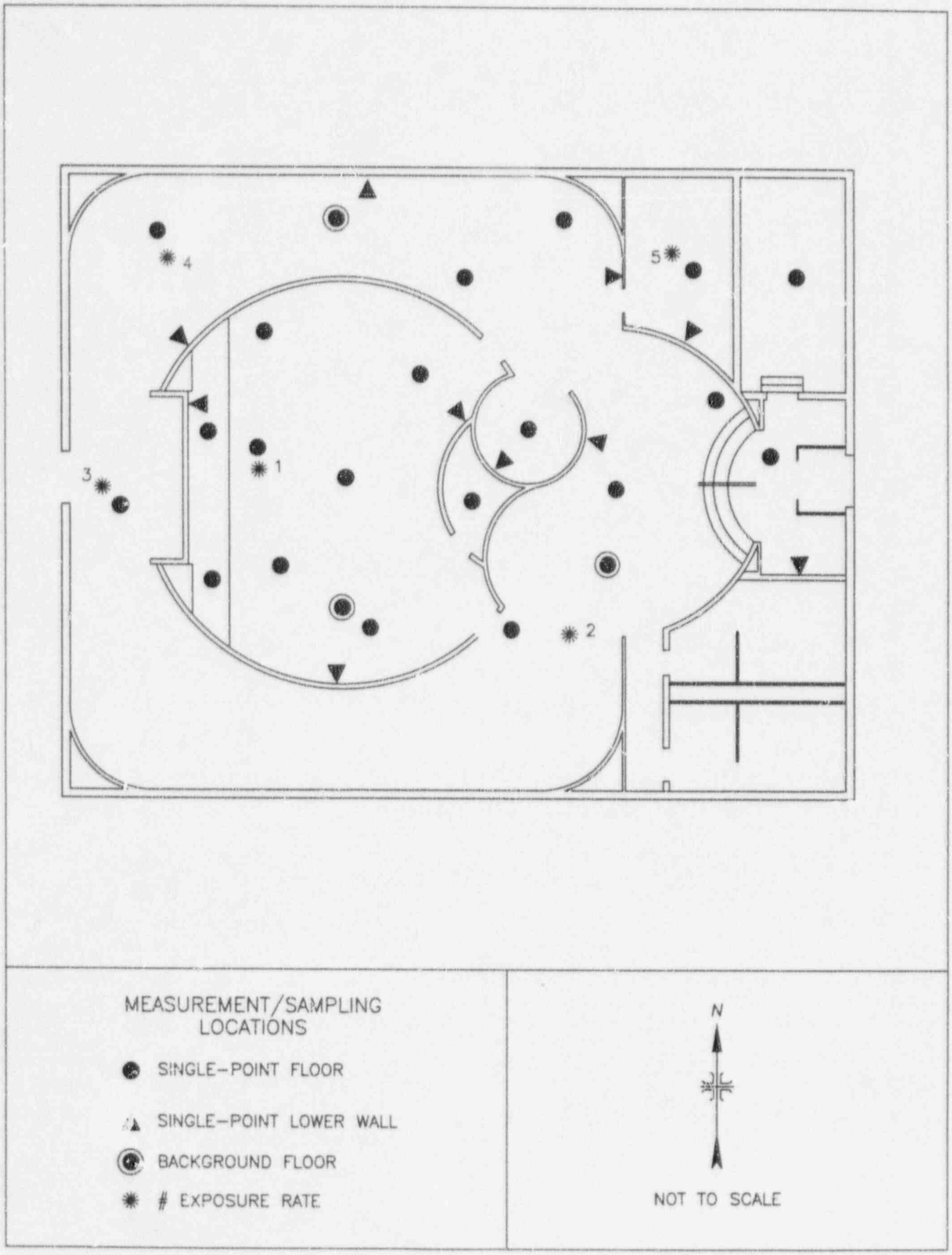

FIGURE 3: A0007, Visitor Center #107 – Measurement and Sampling Locations

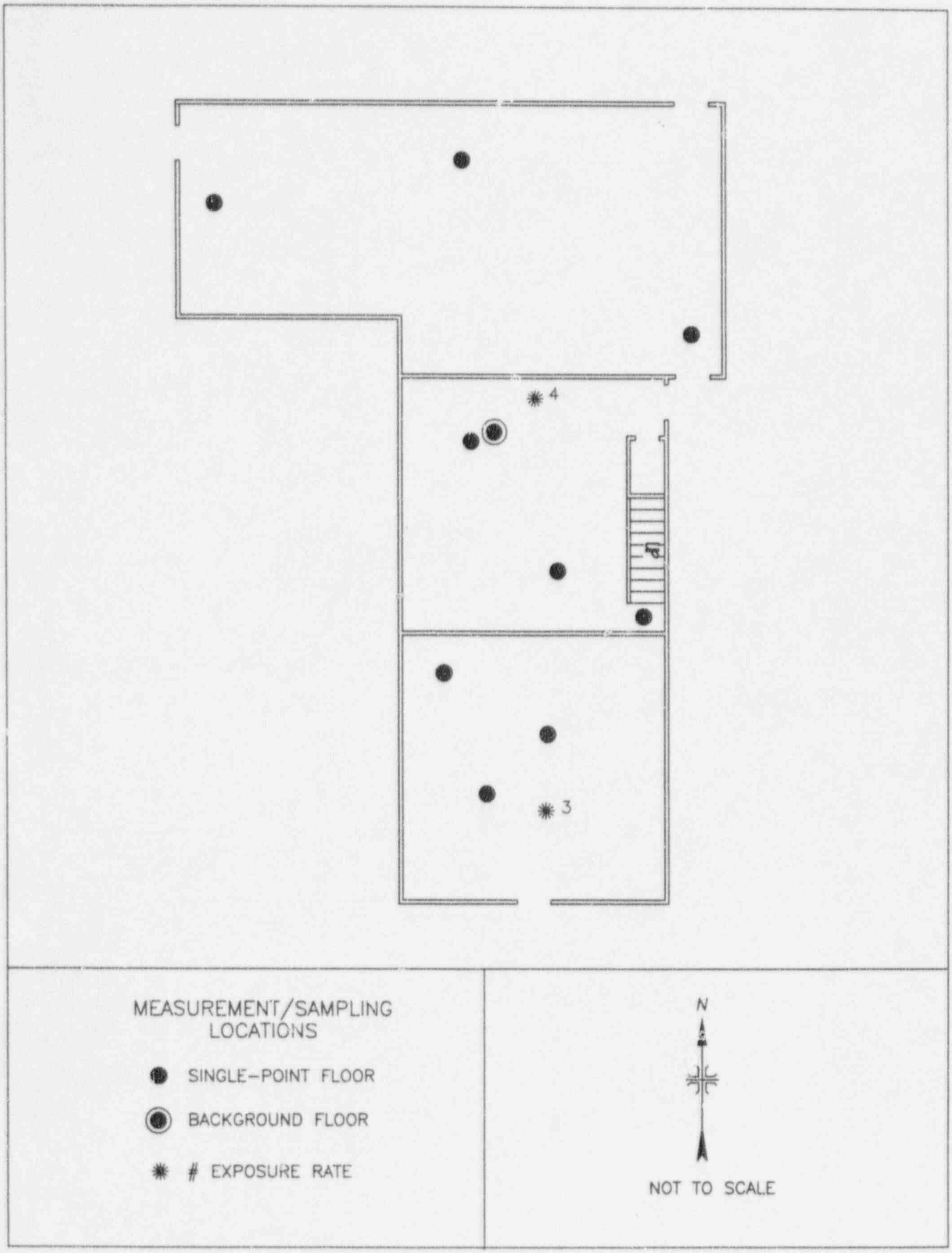

FIGURE 4: B0006, Equipment Building #17 – Measurement and Sampling Locations

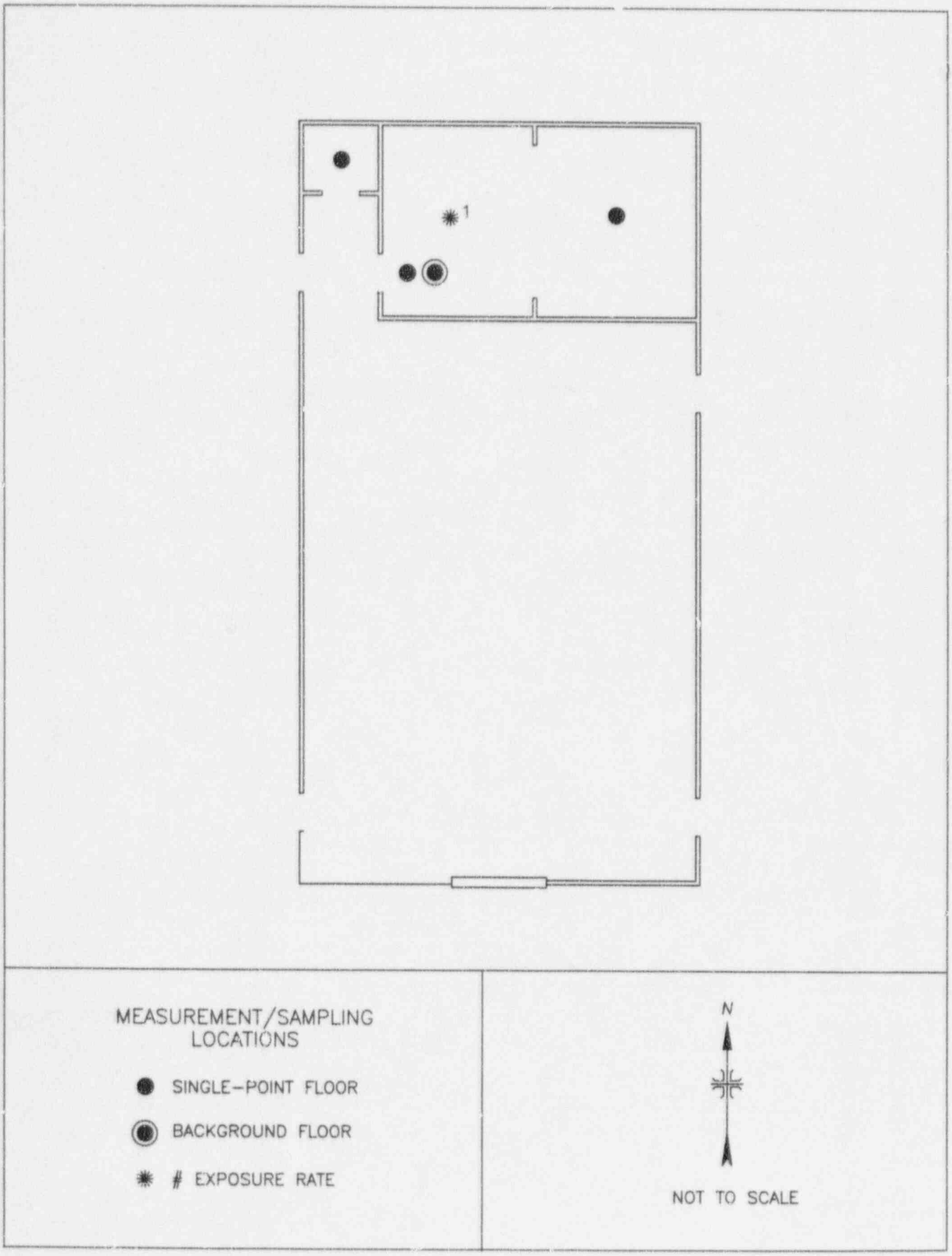

FIGURE 5: B0006, Machine Shop #15 -- Measurement and Sampling Locations

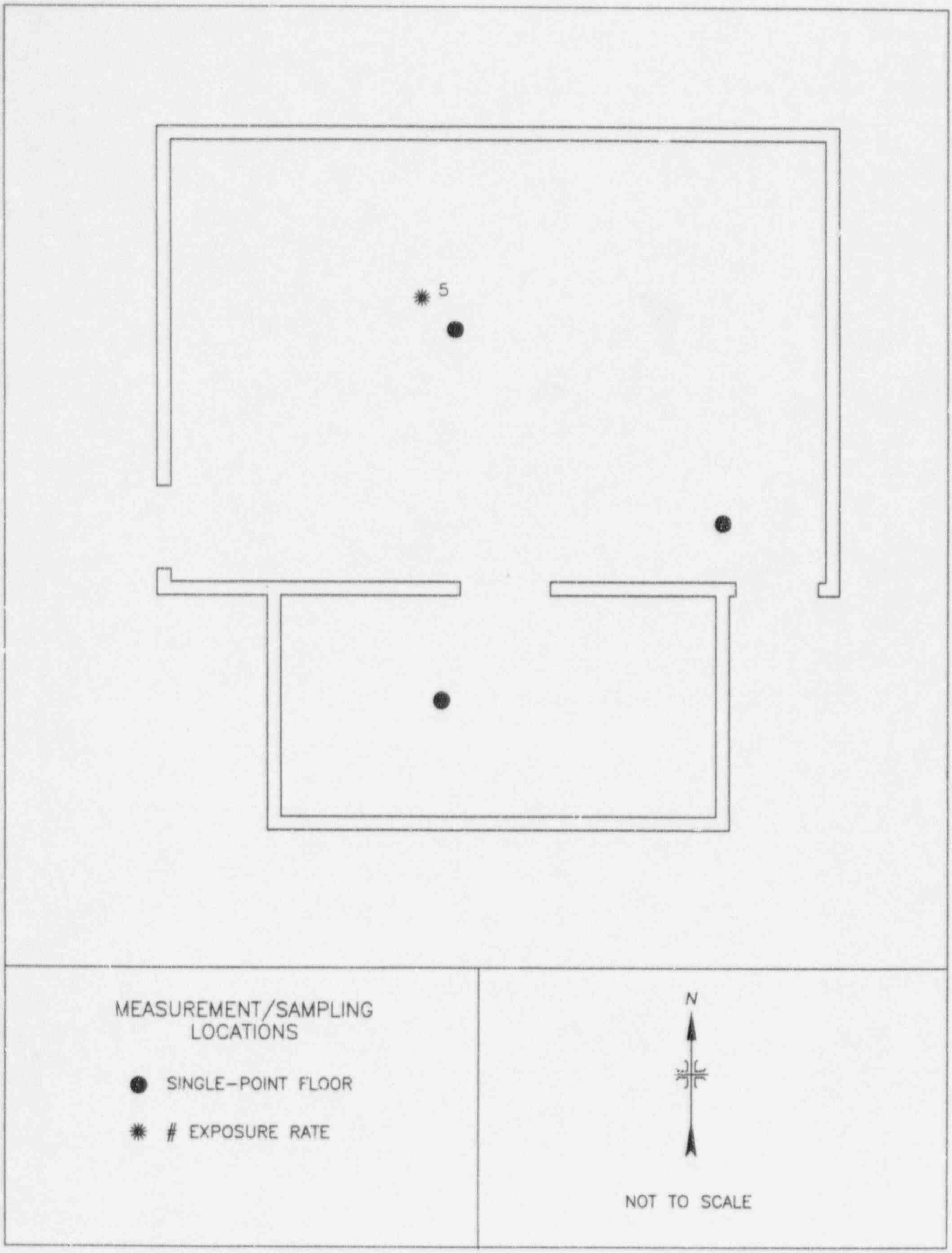

FIGURE 6: B0006, Paint Shed #1003 – Measurement and Sampling Locations

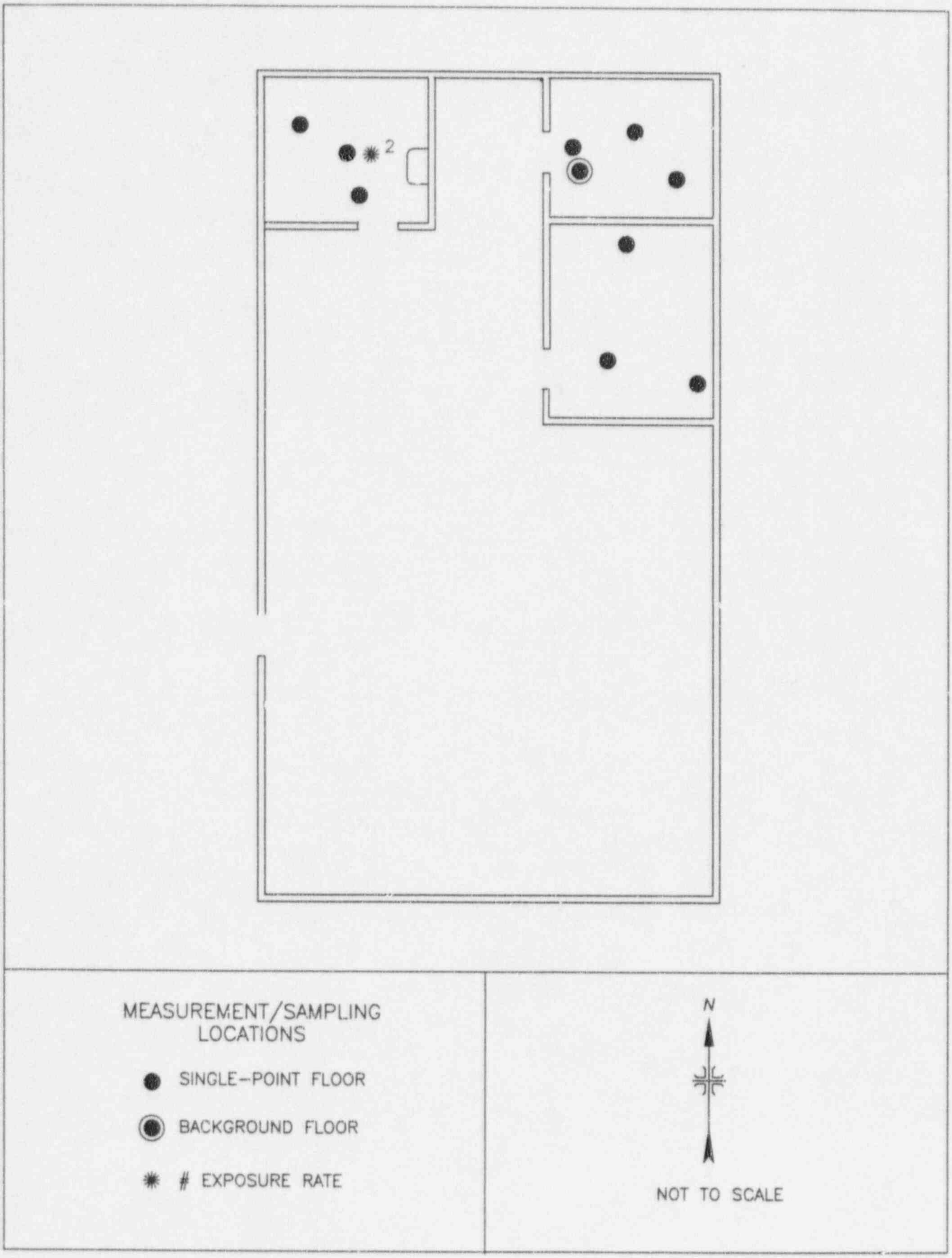

FIGURE 7: B0006, Weld Shop #16 – Measurement and Sampling Locations

621-009 (3)

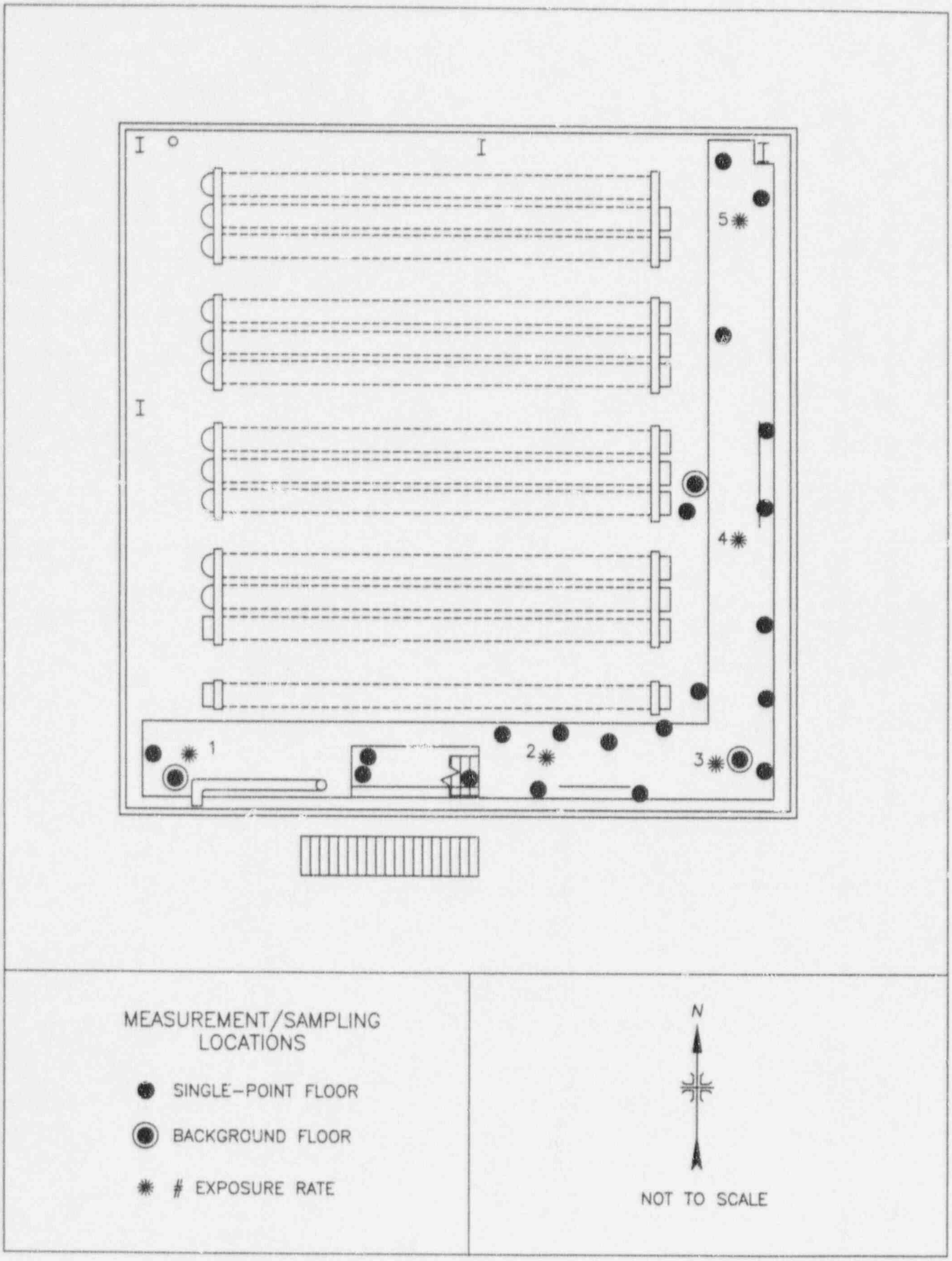

FIGURE 8: B0012, Building 1, Helium Storage – Measurement and Sampling Locations

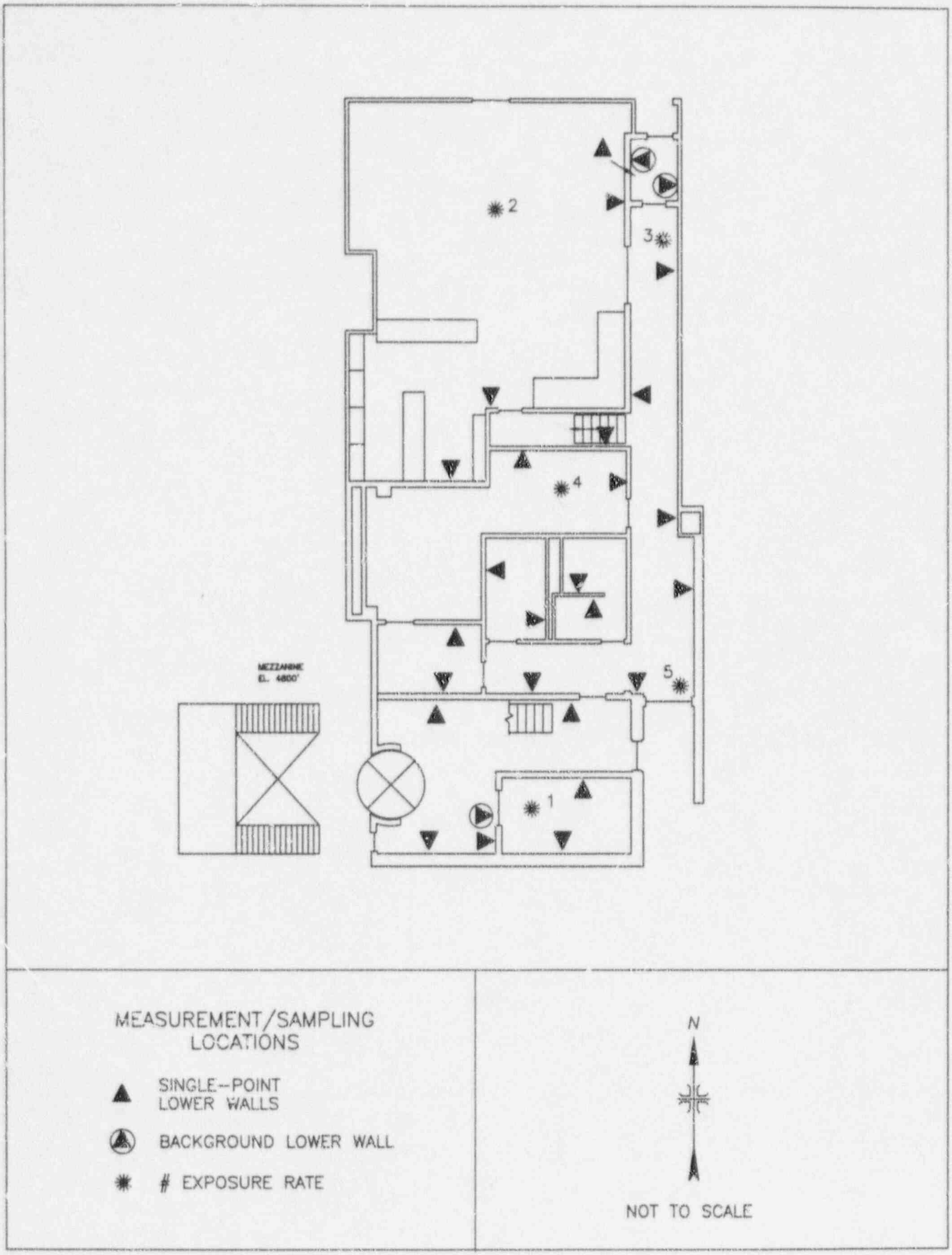

FIGURE 9: C0004, Turbine Building, Northwest Office Areas/Electrical Shop – Measurement and Sampling Locations

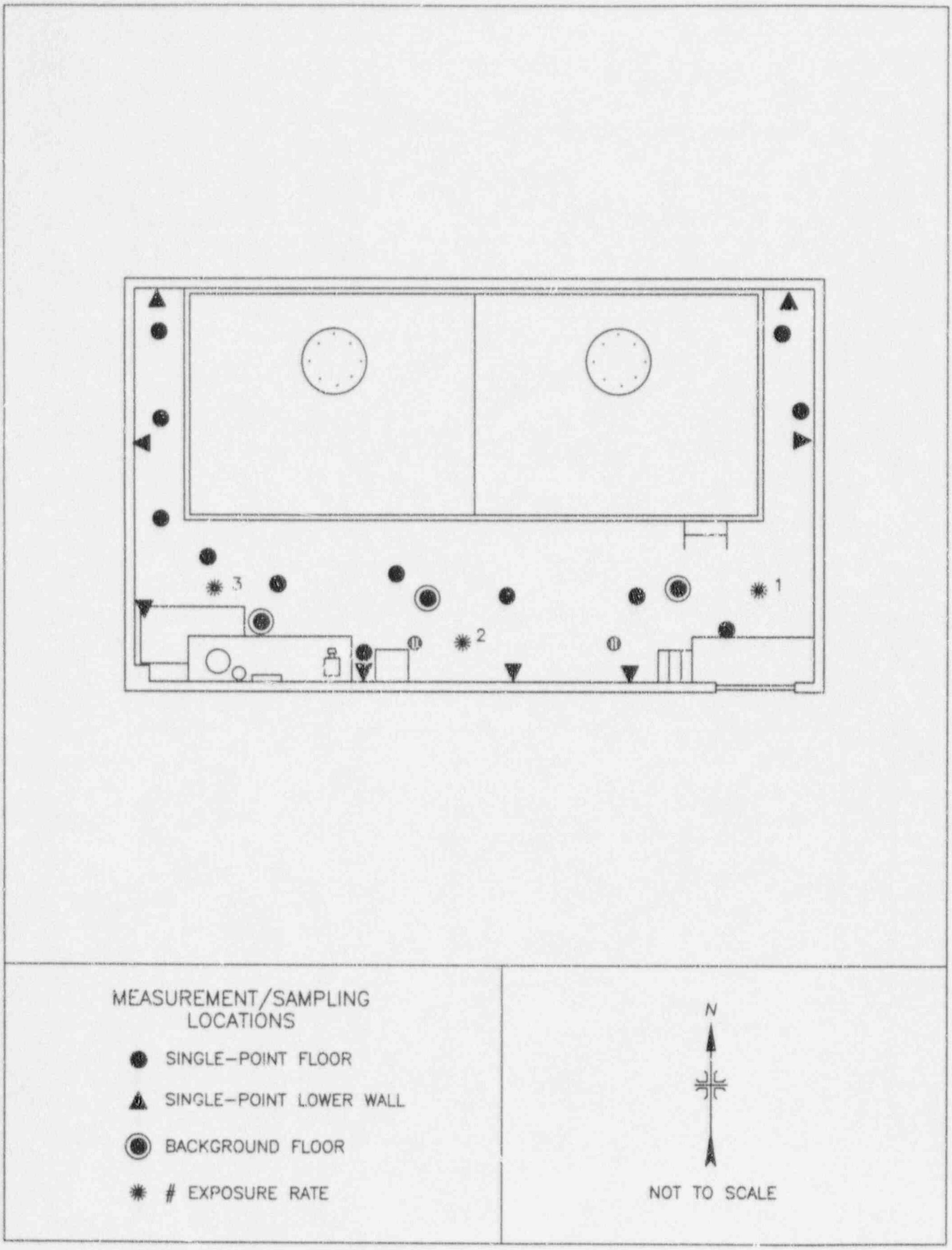

FIGURE 10: C0009, Turbine Lube Oil Storage Tank Room – Measurement and Sampling Locations

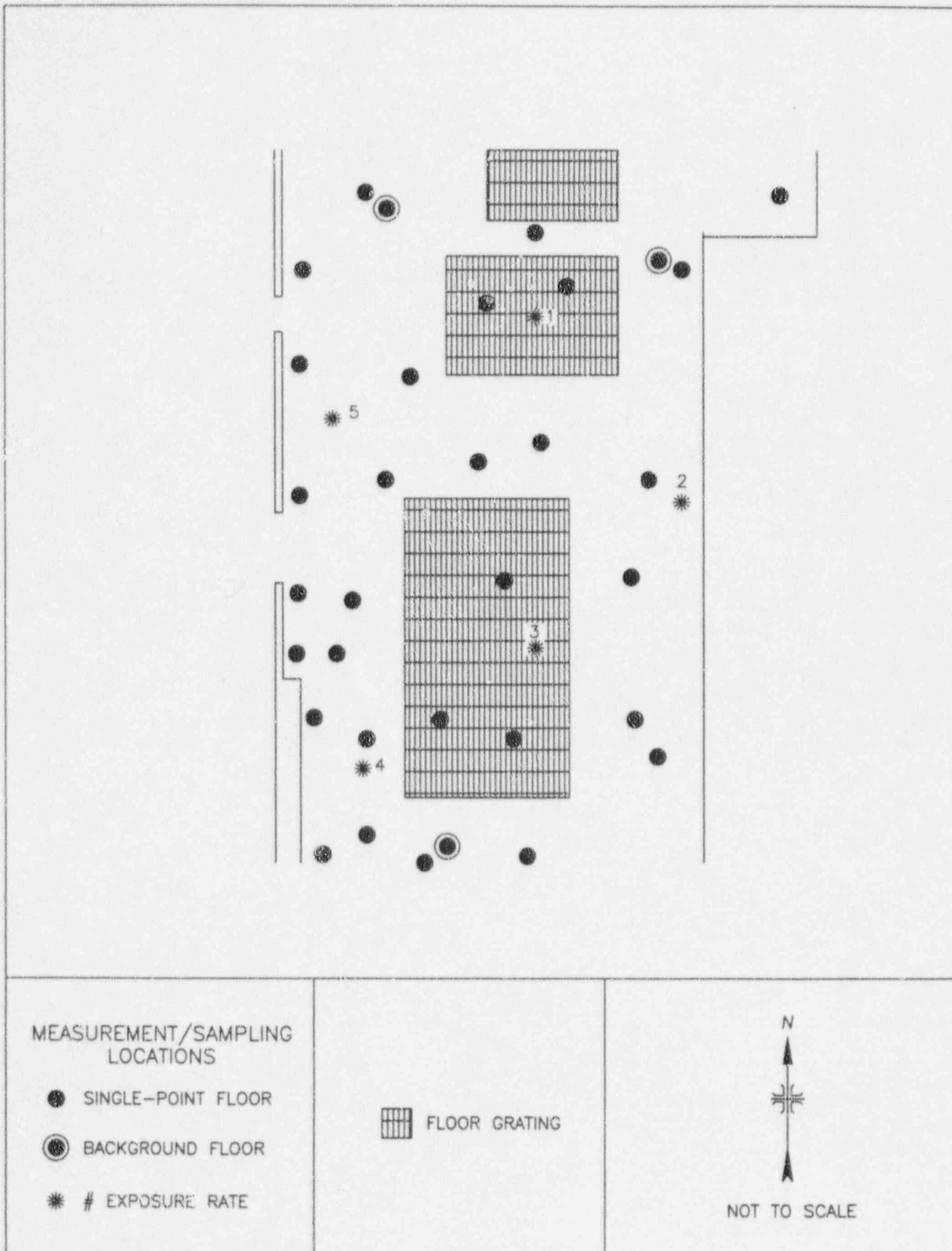

FIGURE 11: C0030, Turbine Deck, Northwest General Area — Measurement and Sampling Location

621-006 (2)

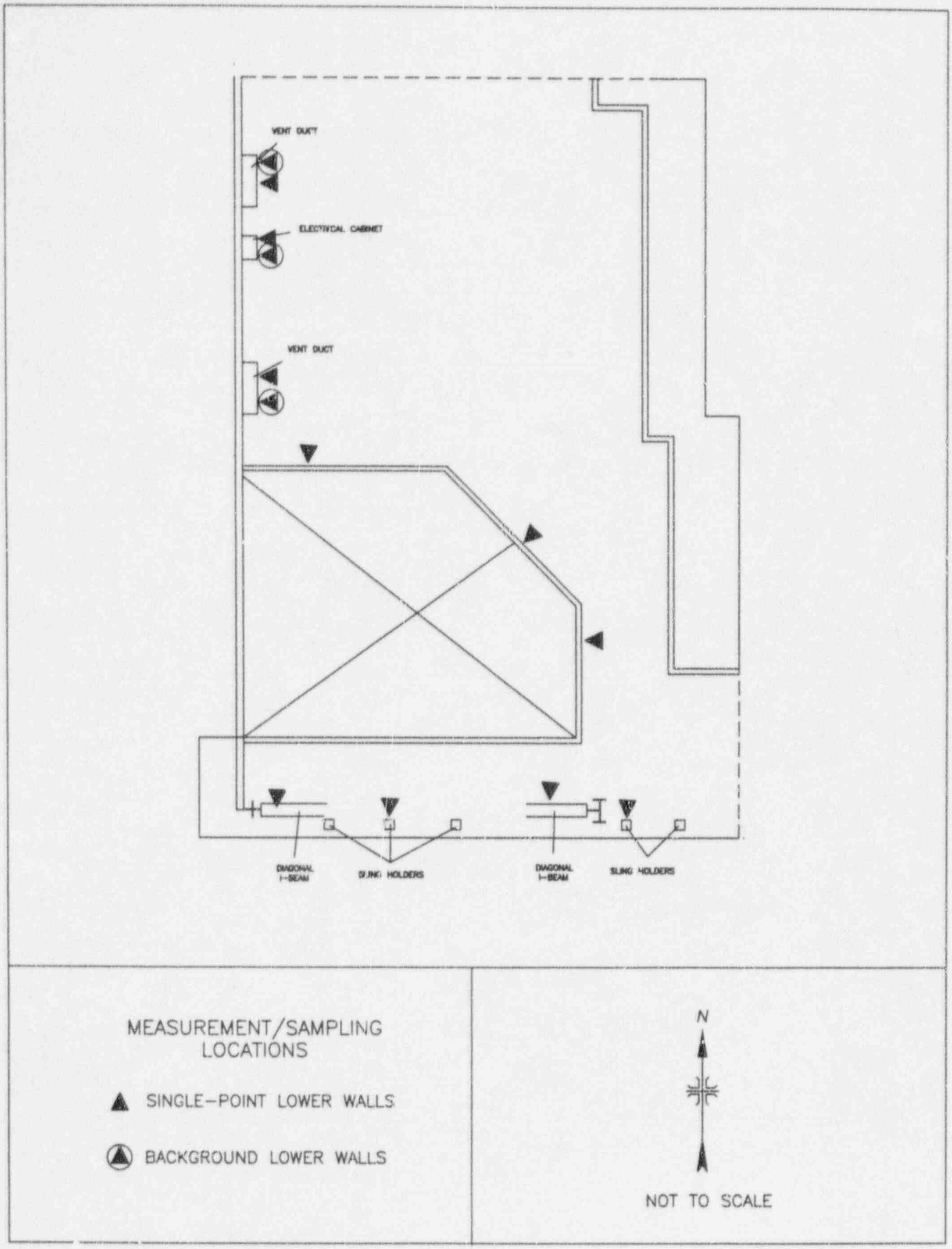

FIGURE 12: C0031, Turbine Deck, Southwest General Area – Measurement and Sampling Locations

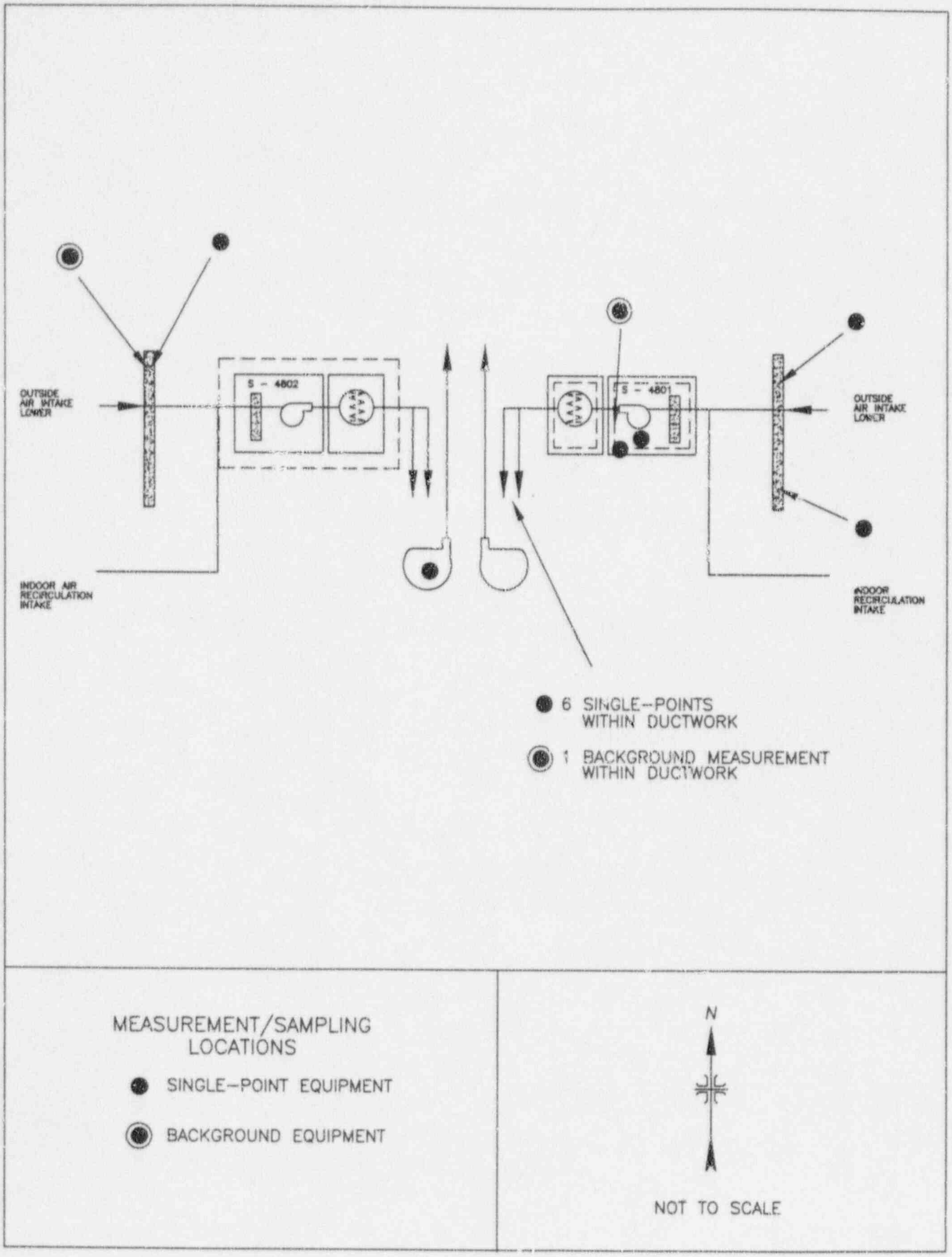

FIGURE 13: D4800, Alternate Cooling System – Measurement and Sampling Locations

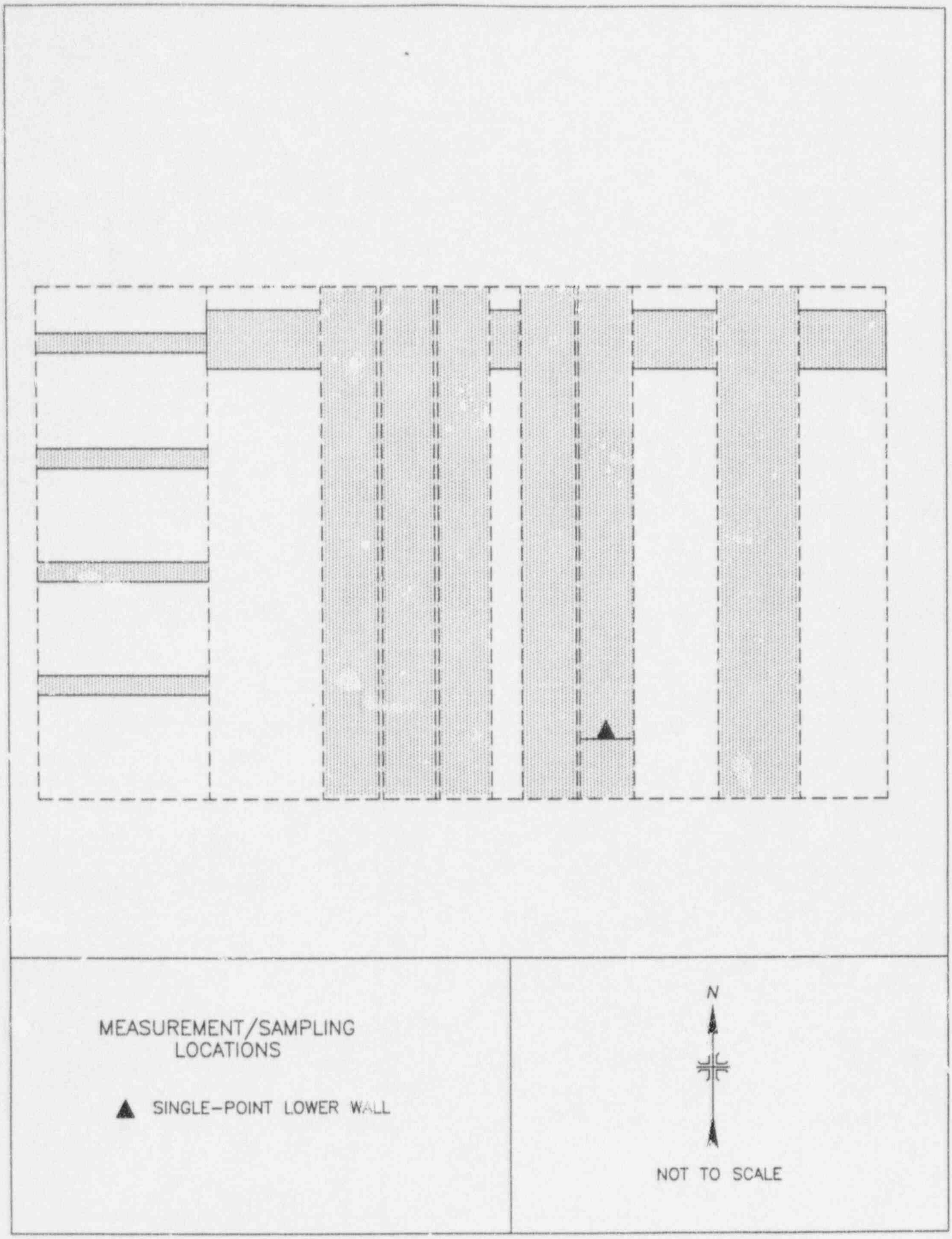

FIGURE 14: F0077, Reactor Building, Level 5 East, SE General Area – Measurement and Sampling Locations

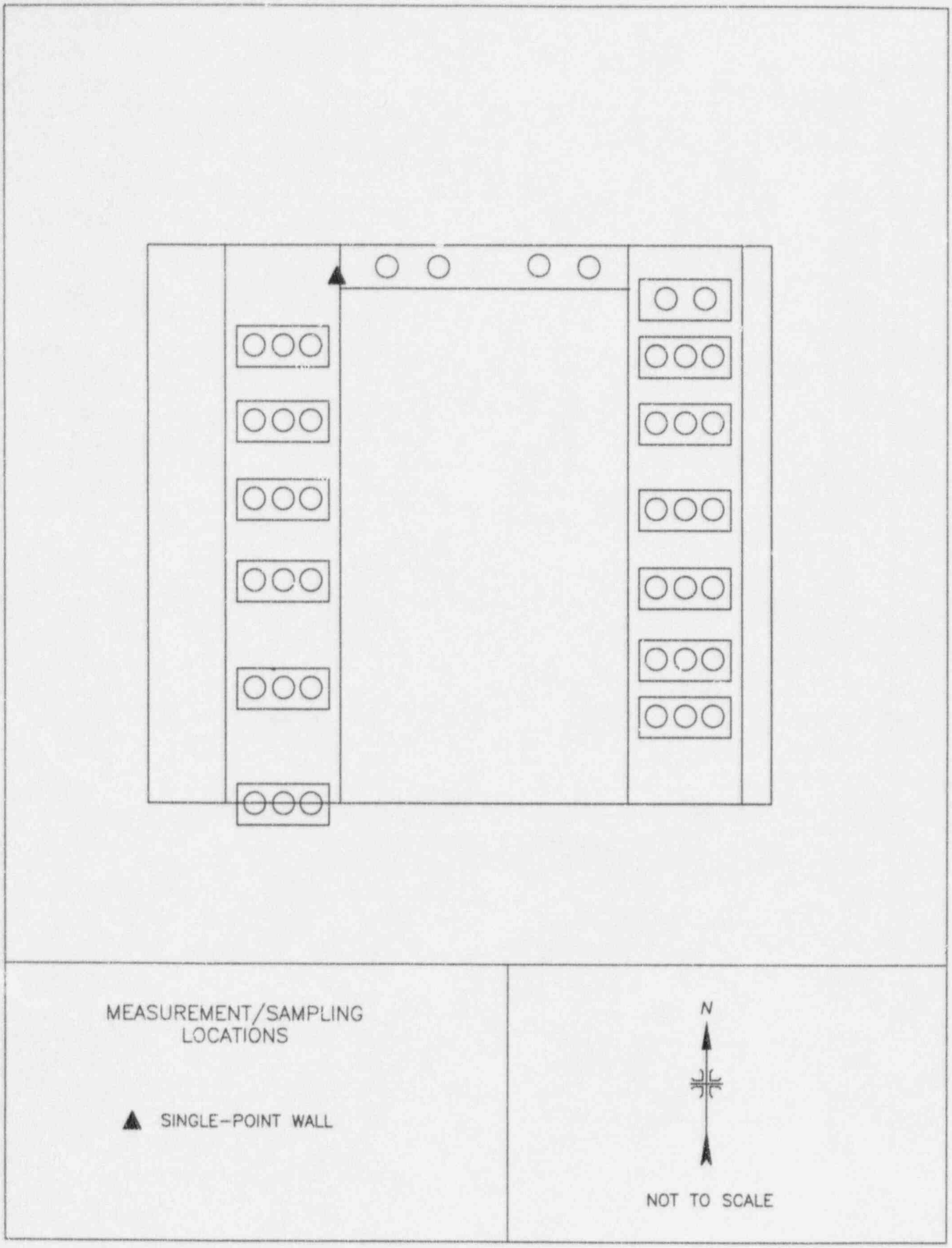

FIGURE 15: F0084, Reactor Building, Level 7 West, SW General Area — Measurement and Sampling Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS
FORT ST. VRAIN NUCLEAR STATION
PLATTEVILLE, COLORADO

Group/Survey Unit	Figure Number(s)	Number of Measurement Locations	Range of Total Activity (dpm/100 cm²)	Range of Removable Activity (dpm/100 cm²)	
			Beta	Alpha	Beta
A0007/BZ003					
Floor and Lower Walls	3	30	-770 to 310	<9	<15
B0006/FZ001					
Floors	4 to 7	24	-300 to 1,100	<9	<15
B0012/FZ002					
Floors	8	20	-370 to -60	<9	<15
C0004/WZ001					
Lower Walls	9	25	-890 to 490	<9	<15
C0009/BZ001					
Floor and Lower Walls	10	20	-460 to 210	<9	<15
C0030/FZ001					
Floors	11	30	-650 to 150	<9	<15
C0031/SZ001					
Structures and Equipment	12	10	-380 to 180	<9	<15
D4800					
Alternate Cooling System	13	12	-480 to 1,200	<9	<15 to 18
DEBRIS FROM BUILDING 28					
Miscellaneous	NA	20	-370 to 1,400	<9	<15
F0077 LEVEL 5 EAST - SE GENERAL AREA (REACTOR BUILDING)					
Wall	14	1	20,000	<9	<15
F0084 LEVEL 7 WEST - SW GENERAL AREA (REACTOR BUILDING)					
Wall (Location in F0115)	15	1	9,200	<9	<15

TABLE 2

COMPARISON OF EXPOSURE RATES
FORT ST. VRAIN NUCLEAR STATION
PLATTEVILLE, COLORADO

Group/Survey Unit Location[a]	Total Exposure Rate (µR/h) at 1 m Above Surface[b]	
A0007/BZ003	**ESSAP**	**FSV**
1	21	19 to 23
2	21	
3	19	
4	20	
5	19	
B0006/FZ001	**ESSAP**	**FSV**
1	14	About 13.5
2	13	
3	16	
4	16	
5	15	
B0012/FZ002	**ESSAP**	**FSV**
1	10	9 to 10
2	9	8 to 10
3	10	5 to 7
4	11	4 to 6
5	10	5 to 6
C0004/WZ001	**ESSAP**	**FSV**
1	17	14 to 18
2	14	
3	14	
4	14	
5	16	

TABLE 2 (Continued)

COMPARISON EXPOSURE RATES
FORT ST. VRAIN NUCLEAR STATION
PLATTEVILLE, COLORADO

Group/Survey Unit Location[a]	Total Exposure Rate (μR/h) at 1 m Above Surface[b]	
C0009/BZ001	**ESSAP**	**FSV**
1	12	About 12
2	12	
3	10	
C0030/FZ001	**ESSAP**	**FSV**
1	10	11 to 14
2	13	
3	11	
4	14	
5	14	

[a]Refer to Figures 3 through 11.

[b]Total exposure rates provided; background has not been subtracted.

TABLE 3

STATISTICAL TEST FOR SURFACE ACTIVITY MEASUREMENTS

Group/Survey Unit	FSV			ESSAP			Test Statistic (t)[a]	Critical Value of t-test
	No. Meas.	Mean (dpm/100 cm^2)	Standard Deviation (dpm/100 cm^2)	No. Meas.	Mean (dpm/100 cm^2)	Standard Deviation (dpm/100 cm^2)		
A0007/BZ003	67	31.5	365.3	30	-124.4	241.6	-2.14	1.66
B0006/FZ001	36	325.3	349.8	24	26.3	305.0	-3.41	1.67
B0012/FZ002	78	182.9	155.8	20	-203.0	91.3	-10.6	1.66
C0004/WZ001	110	72.9	368	25	-10.0	252.8	-1.07	1.66
C0009/BZ001	87	-27.3	257.9	20	-123.8	220.5	-1.55	1.66
C0030/FZ001	314	-99.6	233.8	30	-144.2	240.3	-1.00	1.65
C0031/SZ001	32	397.1	291	10	-151.4	171.1	-5.63	1.68
D4800	57	180	303.7	12	167.7	449.2	-0.12	1.67

[a]Null hypothesis is stated as $\mu_{FSV} \geq \mu_{ESSAP}$. The null hypothesis is accepted at the 95% confidence level if the test statistic (t) is less than the critical value.

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Oak Ridge Institute for Science and Education. Final Report—ORISE Support of NRC License Inspection at Fort St. Vrain on January 22 to 25, 1996. Oak Ridge, TN; March 1996b.

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U.S. Nuclear Regulatory Commission (NRC). Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86. Washington, D.C.; June 1974.

U.S. Nuclear Regulatory Commission. Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities. NUREG-0586; August 1988.

U.S. Nuclear Regulatory Commission (letter from C.L. Pittiglio), to A.J. Bortz, Long Island Power Authority, subject "Approval of a Modification of Facility Release Criteria for Tritium and Iron-55 Surface Contamination at Shoreham Nuclear Power Station, Unit 1," June 7, 1994.

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 author or his employer.

DIRECT RADIATION MEASUREMENT

Instruments

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

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

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

Detectors

Eberline GM Detector
Model HP-260
Effective Area, 20 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, 126 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

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 detectors slowly over the surface; the distance between the detector 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 (20 cm^2 or 126 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
Beta	-	gas proportional detector with ratemeter-scaler
	-	GM detector with ratemeter-scaler
Gamma	-	NaI scintillation detector with ratemeter

Surface Activity Measurements

Measurements of total beta activity levels were primarily performed using gas proportional and 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 total background level at each direct measurement location was obtained by adding the construction material background and the local area background. The local

area background was obtained in each surveyed area by placing the licensee's plastic shield between the surface being assessed and the detector. The detector's response with the plastic shield in place was assumed to be only from the local area gamma radiation levels, and not from the surface material. Because different building construction materials (poured concrete, concrete block, metal, wood, etc.) can have very different background levels, average background counts were determined for each material encountered in the surveyed area at a location of similar construction and having no known radiological history. The construction material background count rates for the proportional detectors averaged 268 cpm for red brick, 228 cpm for brick, 114 cpm for painted concrete, 594 cpm for tile brick, 264 cpm for concrete floor, 73 cpm for carpet, and 31 cpm on metal. The local area background count rates ranged from 334 to 594 cpm. Net count rates were determined by subtracting the appropriate material background and local area background from the gross count rate for each measurement location. Beta efficiency factors ranged from 0.25 to 0.26 for the gas proportional detectors and from 0.18 to 0.19 for the GM detectors calibrated to Tc-99. The physical window areas for the gas proportional and the GM detectors were 126 cm^2 and 20 cm^2, respectively.

Removable Activity Measurements

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

Exposure Rate Measurements

Measurements of gamma exposure rates were performed at 1 m above the surface, using a pressurized ionization chamber (PIC).

Analytical Procedures

Gross Alpha/Beta

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

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 concentration (MDC), were based on 2.71 plus 4.65 times the standard deviation of the background count [$2.71 + 4.65\sqrt{BKG}$]. When the activity was determined to be less than the MDC of the measurement procedure, the result was reported as less than MDC. 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

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

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 9 (April 1995)
- Laboratory Procedures Manual, Revision 9 (January 1995)
- Quality Assurance Manual, Revision 7 (January 1995)

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.

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

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 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 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

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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 contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement should be assumed to be contaminated in excess of the 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.