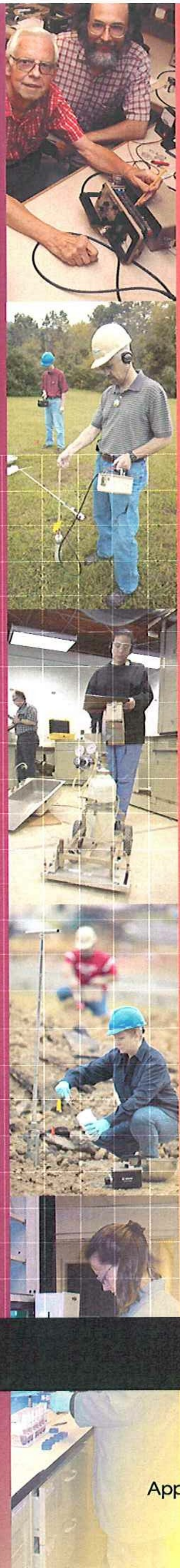


CONFIRMATORY SURVEY REPORT FOR THE QUEHANNA DECOMMISSIONING PROJECT KARTHAUS, PENNSYLVANIA

W. C. Adams

Prepared for the
Office of Federal & State Materials &
Environmental Management Program
U.S. Nuclear Regulatory Commission

ORISE
Oak Ridge Institute for Science and Education

ORISE 07-IEAV-0988

CONFIRMATORY SURVEY REPORT FOR THE QUEHANNA DECOMMISSIONING PROJECT KARTHAUS, PENNSYLVANIA

Prepared by

W. C. Adams

Oak Ridge Institute for Science and Education
Oak Ridge, Tennessee 37831-0117

Prepared for the
Office of Federal and State Materials and Environmental Management Programs
U.S. Nuclear Regulatory Commission

FINAL REPORT

AUGUST 2007

This report is based on work performed under an Interagency Agreement (NRC Fin. No. F1008) between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy. The Oak Ridge Institute for Science and Education performs complementary work under a contract with the U.S. Department of Energy.

CONFIRMATORY SURVEY REPORT FOR THE QUEHANNA DECOMMISSIONING PROJECT KARTHAUS, PENNSYLVANIA

Prepared by: Wade C. Adams Date: 8/3/2007
W. C. Adams, Project Leader

Reviewed by: Sarah Roberts Date: 8/3/07
S. J. Roberts, Survey Projects Manager

Reviewed by: R.D. Condra Date: 8/6/07
R. D. Condra, Laboratory Manager

Reviewed by: Ann T Payne Date: 8/7/07
A. T. Payne, Quality Manager

Reviewed by: E. W. Abelquist Date: 8/6/07
E. W. Abelquist, Vice-President & Director
Independent Environmental Assessment and Verification

ACKNOWLEDGMENTS

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

FIELD STAFF

T. L. Brown
T. D. Herrera
E. Montalvo

LABORATORY STAFF

R. D. Condra
W. P. Ivey
J. S. Cox
W. F. Smith

CLERICAL STAFF

A. Ramsey
K. M. Moore

ILLUSTRATOR

T. L. Brown
T. D. Herrera

TABLE OF CONTENTS

PAGE

LIST OF FIGURES

PAGE

LIST OF TABLES

PAGE

ABBREVIATIONS AND ACRONYMS

b_i	number of background counts in the interval
d′	index of sensitivity
ε_i	instrument efficiency
ε_s	surface efficiency
ε_{total}	total efficiency
AEC	U.S. Atomic Energy Commission
ARC	Atlantic-Richfield Corporation
BKG	background
CFR	Code of Federal Regulations
cm	centimeter
cm^2	square centimeters
Co-60	cobalt-60
cpm	counts per minute
Cs-137	cesium-137
CWC	Curtis-Wright Corporation
D&D	decontamination and decommissioning
DCGL	derived concentration guideline level
DCNR	Department of Conservation and Natural Resources
DOE	U.S. Department of Energy
DP	decommissioning plan
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
ESL	Energy*Solutions*, LLC
FA	Finishing Area
FSS	final status survey
FSSP	final status survey plan
FSSR	final status survey report
GM	Geiger-Muller
ha	hectare
ISM	integrated safety management
ISO	International Organization for Standardization
ITP	Intercomparison Testing Program
keV	kiloelectron volt
km	kilometers
LLRW	low-level radioactive waste
m	meter
MAPEP	Mixed Analyte Performance Evaluation Program
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
MDCR	minimum detectable count rate
MeV	million electron volts
MMC	Martin Marietta Corporation
min	minute
mg/cm^2	milligrams per square centimeter
mrem/y	millirem per year

ABBREVIATIONS AND ACRONYMS (continued)

NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NRIP	NIST Radiochemistry Intercomparison Program
NUMEC	Nuclear Materials and Equipment Corporation
ORISE	Oak Ridge Institute for Science and Education
PADEP	Pennsylvania Department of Environmental Protection
pCi/g	picocuries per gram
PSU	Pennsylvania State University
QDP	Quehanna Decommissioning Project
ROC	radionuclides-of-concern
s	second
SNAP	Systems for Nuclear Auxiliary Power
Sr-90	stronitum-90
$SrTiO_3$	strontium titanate
STI	Scientech, Incorporated
SU	survey unit
TAP	total absorption peak
WWTB	Waste Water Treatment Building

CONFIRMATORY SURVEY REPORT FOR THE QUEHANNA DECOMMISSIONING PROJECT KARTHAUS, PENNSYLVANIA

INTRODUCTION AND SITE HISTORY

In 1957, the Curtis-Wright Corporation (CWC) finished construction of a jet engine and nuclear research facility at the Quehanna Site located in Karthaus, Pennsylvania. Following the construction of the facility, the U.S. Atomic Energy Commission (AEC), a precursor to the U.S. Nuclear Regulatory Commission (NRC), issued a license to CWC in 1958 to operate a swimming pool research reactor. The license also included the use of hot cells, laboratories, and support features (STI 2004).

In September 1960, CWC donated the facility and land to Pennsylvania State University (PSU) which subsequently leased the hot cells to Martin Marietta Corporation (MMC). In 1962, MMC used the hot cells to manufacture several prototype thermoelectric generators, known as Systems for Nuclear Auxiliary Power (SNAP) generators, for the AEC. These power sources, which were designed to furnish power for remotely operated, automatically reporting weather stations, navigation buoys, etc., contained very high specific activity strontium-90 (Sr-90) in the form of strontium titanate ($SrTiO_3$). MMC's radioactive material license allowed them to maintain megacurie quantities of Sr-90. When MMC terminated its lease in 1967, they partially decontaminated the facility. However, licensable quantities of Sr-90 remained behind as structural contamination. MMC was the last licensee to use Sr-90 at the Quehanna Site (STI 2004).

In 1967, PSU gave its interest in the Quehanna Site back to the Commonwealth of Pennsylvania which in turn leased the facility to the Nuclear Materials and Equipment Corporation (NUMEC), a subsidiary of the Atlantic-Richfield Corporation (ARC). NUMEC used the reactor pool to hold a large cobalt-60 (Co-60) irradiator containing in excess of 1 million curies of Co-60 for projects involving food irradiation and irradiation of polymer-impregnated hardwood, and other applications of intense gamma radiation. In 1978, a group of ARC employees bought the wood irradiation process at the Quehanna Site, including the Co-60 irradiator and related equipment. The new company, PermaGrain, was issued Radioactive Materials License Number 37-17860-01 by the NRC for the irradiator and also assumed "caretaker" responsibilities for the material left behind by previous tenants (STI 2004).

The Pennsylvania Department of Environmental Protection (PADEP) assumed the official radioactive materials license since PermaGrain filed for bankruptcy in December 2002. PADEP renewed the license in September 2003 under NRC Radioactive Materials License Number 37-17860-02. Currently, the Commonwealth of Pennsylvania owns the Quehanna Site and the surrounding real estate and the Pennsylvania Department of Conservation and Natural Resources (DCNR) Bureau of Forestry administers the land.

The contaminants of concern at the Quehanna Site are Sr-90 with possible residual Co-60 from the use of and manufacture of cobalt irradiators. However, measurable quantities of Co-60 are not expected since extensive remediation has taken place in the localized areas where Co-60 was known to exist. There is also a small potential for activation products from operations of the test reactor (STI 2004).

The original objective of the decontamination and decommissioning (D&D) project was to decontaminate and free-release the entire Quehanna Site for reuse for industrial purposes by the existing tenant, and to terminate NRC Radioactive Materials License Number 37-17860-02. The initial Quehanna Decommissioning Plan (DP), prepared by Scientech, Inc. (STI), was prepared based on the requirements of NRC Regulatory Guide 1.86 (STI 2004). STI's decommissioning activities included: 1) the removal of the Hot Cell 4 process system by the use of a remotely controlled robot, 2) the removal of the Co-60 irradiator sources from the reactor pool and hot cells, 3) decontamination of areas such as the laboratories, production and storage areas, and offices, 4) surveys and demolishing of interior structures north of the reactor bay and cell face (e.g. walls, ceiling and floor tiles, etc.), and 5) the disposal of debris as clean waste or low-level radioactive waste [LLRW (STI 2004)].

STI performed final status surveys (FSS) on the site and submitted a final status survey report (FSSR) on the FSS findings and submitted the report to the NRC (ESL 2005) for review and approval. Based on this FSSR, the NRC Headquarters and Region I Offices requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory surveys at the Quehanna Site. The initial confirmatory surveys were conducted during the periods of November 8 through 10, 2004 and May 3 through 4, 2005. The previous confirmatory survey activities for the formerly classified Class 1 and Class 2 interior building areas failed to confirm that the radiological conditions at the Quehanna Decommissioning Project (QDP) met the approved unrestricted release limits specified in the original DP (ESL 2003). Beta surface scans during the previous survey activities identified several areas of elevated activity; 66 of the 120 direct measurements collected during the previous survey activities exceeded the maximum criterion of 3,000 disintegrations per minute per 100 square centimeters (dpm/100 cm^2) for Sr-90 with the beta surface activities ranging from -275 to 182,800 dpm/100 cm^2. Removable beta activity ranged from -5 to 178 dpm/100 cm^2 (ORISE 2005a and b).

Subsequently, the decommissioning contractor, Energy*Solutions*, LLC (ESL), formerly STI, issued a revised DP with dose-based release criteria replacing the surface contamination guidelines taken from Regulatory Guide 1.86 that were specified in the previous version of the DP (ESL 2006a). The DP was revised and submitted to the NRC in March of 2006. The revisions were based on the fact that: 1) ORISE identified areas of elevated activity above the unrestricted release guidelines during the confirmatory survey activities performed in May of 2005; and 2) the end use of the site changed when the existing tenant declared bankruptcy and vacated the site. The current plan is to designate the site property as a "Wild Area". The revised approach followed the requirements of Title 10, Code of Federal Regulations (CFR), Part 20, Subpart E which specifies that the unrestricted release of a site shall assure that the average member of the critical group shall receive no more than 25 millirem per year (mrem/y) after the site has been closed and the license terminated (ESL 2006a). The NRC issued a license amendment needed to approve the revised DP.

The revised FSS approach is based on the guidance of the Multi-Agency Radiation Survey and Site Investigation Manual [MARSSIM (NRC 2000)], and specifies the requirements for structural surface surveys, concrete core samples, and surface and subsurface soil sampling (ESL 2006b).

Currently, the site decommissioning contractor, ESL, has performed FSS of the Quehanna Site based on a NRC-approved revised final status survey plan [FSSP (ESL 2006b)]. The objective of the FSSP was to demonstrate that the radiological conditions at the Quehanna Site satisfy the release criteria specified in the revised DP so that the site can be released for unrestricted use (ESL 2006a).

Regulators that are involved in the D&D project include the NRC, the DCNR, and PADEP. The PADEP maintains the license for the site. Based on the updated FSS requirements, the NRC's Office of Federal and State Materials and Environmental Management Programs and the Region I

Office requested that ORISE perform additional confirmatory surveys of the Quehanna Site in Karthaus, Pennsylvania.

SITE DESCRIPTION

The Quehanna Site is located at 115 Reactor Road, Karthaus, Clearfield County, Pennsylvania (Figure 1). The site is approximately 35 kilometers [km (21 miles)] northeast of Clearfield, Pennsylvania and is located in the 20,000-hectare [ha (50,000-acre)] Quehanna Wild Area of the Moshannon State Forest. The area is heavily wooded and sparsely populated. The Quehanna Facility has a basement, main and second floor area of approximately 3,700 meters [m (40,000 square feet)].

The Quehanna Site includes or included many affected structures and systems, such as the hot cells complex (Cell Structure), the Waste Water Treatment Building (WWTB) with associated underground tanks and piping, the Reactor Bay, and the hot cell ventilation system. Some of these systems and structures have been removed as clean debris or partially decontaminated and disposed of as LLRW. The facility also includes other laboratories, production areas, storage areas, and offices formerly used by the previous licensee, PermaGrain. The Quehanna site and facility plot plans are provided in Figures 2 and 3.

OBJECTIVES

The objectives of the confirmatory survey were to provide independent field data reviews and to generate independent radiological data for use by the NRC in evaluating the adequacy and accuracy of the licensee's procedures and FSS data. Additionally, this review provided assurance that the licensee adequately designed the FSS and fulfilled the commitments contained in the DP.

DOCUMENT REVIEW

ORISE has reviewed ESL's revised DP and revised FSSP for adequacy and appropriateness taking into account commitments contained in these documents that were approved by the NRC (ESL 2006a and b). These documents contain the release criteria for the site, along with the documentation on the derivation of the release criteria. The final survey data for the survey units (SU) to be evaluated were reviewed by ORISE prior to mobilization to the site and while at the site during confirmatory survey activities. ORISE reviewed and evaluated the radiological data, in accordance with the ORISE survey plan and other referenced documents, to ensure that FSS procedures and results adequately met site DP and FSSP commitments.

PROCEDURES

ORISE visited the Quehanna Site and performed visual inspections and surface activity measurements. The confirmatory survey activities, performed on December 5 and 6, 2006, were conducted in accordance with a site-specific survey plan and with the ORISE Survey Procedures and Quality Program Manuals (ORISE 2006a, 2006b and 2007).

The following radiological survey procedures were used by ORISE to conduct confirmatory surveys of the QDP facility above grade structural surfaces. ORISE selected 16 of the SUs from Table 2-2 in the FSSP for which ESL had provided FSS data for confirmatory surveys. The SUs were selected based on FSS results and previous ORISE site radiological survey results which indicated the presence of discrete Sr-90 particles throughout the main floor portions of the facility during the previous ORISE confirmatory survey activities (ORISE 2005b).

Since the above grade structures, excluding the floor, will be disposed of as LLRW, at the request of the NRC site representative, ORISE performed confirmatory surveys on the lower and upper walls of the main floor. In addition to the confirmatory surveys on the above grade structural surfaces, ORISE performed beta surfaces scans on the main floor surfaces since the majority of the contamination found during the ORISE 2005 confirmatory surveys was identified on the floor. ORISE's previous confirmatory survey results corroborated ESL's subsequent findings that a recontamination event had occurred which affected the entire interior footprint of the structure with the heaviest concentration of contamination being found on the floors of the former Decon and Chem Lab Rooms. To a much lesser extent, contamination was found by ORISE in the Admin Area, Reactor Bay, and Finishing Area (ESL 2006b).

ORISE also performed beta and gamma scans on the floors and lower walls of the basement level Storage and Pump Rooms. ORISE did not perform surface activity measurements in the basement level areas since previous and present ORISE confirmatory surveys did not identify residual surface contamination in those areas (ORISE 2005a and b).

SURVEY UNIT CLASSIFICATION

ESL surveyed all above-grade structures in accordance with MARSSIM. All above-grade structures were classified as Class 3 SUs since the levels of residual radioactivity in these areas exist at a small fraction of the revised release criteria (ESL 2006a). ESL stated that if any areas demonstrated removable activity greater than the removable criteria, those areas would be decontaminated, reclassified, and surveyed as Class 1 SUs; however, no FSS removable activity data exceeded the removable release criteria (ESL 2006b).

REFERENCE SYSTEM

Direct measurement locations were referenced to prominent building features and recorded on SU figures prepared by ESL.

SURFACE SCANS

ORISE performed beta and gamma radiation surface scans within each of the SUs selected for confirmatory surveys. The percentages of scan coverage for each SU selected for confirmatory surveys are presented in Table 1 below:

TABLE 1: SURVEY UNIT SCAN PERCENTAGES

Survey Unit Scan Percentages				
Survey Unit	**Floor Gamma Scan Percentage (%)**	**Beta Scan Percentage (%)**		
		Floor	**Lower Walls (<2 m)**	**Upper Walls (>2 m)**
Mezzanine	75	25	50	--[a]
Service Area	75	50	50	5
Decon Room	75	50	50	--
Chem Lab	75	50	50	--
Vestibule	75	50	50	--
Admin Area	75	50	50	5
Reactor Bay	75	50	50	5
Boiler Room	75	50	50	--
Area Near Old Dock	75	50	50	--
Hydroblast Area	75	50	50	--
Finishing Area (FA)	75	50	50	--
FA Electrical Room	75	50	25	--

TABLE 1: SURVEY UNIT SCAN PERCENTAGES (continued)

Survey Unit Scan Percentages				
Survey Unit	**Floor Gamma Scan Percentage (%)**	**Beta Scan Percentage (%)**		
		Floor	**Lower Walls (<2 m)**	**Upper Walls (>2 m)**
FA Bunker	75	50	25	--
FA Office	75	50	25	--
FA Tool Crib	75	50	25	--
WWTB	75	50	25	--
Storage Room	75	50	25	--
Pump Room	75	50	25	--

[a]Measurement not performed.

During the surface scans, particular attention was given to cracks and joints where material may have accumulated. Scans were performed using Geiger-Muller (GM), hand-held gas proportional and sodium iodide (NaI) scintillation detectors coupled to ratemeters or ratemeter-scalers with audible indicators.

SURFACE ACTIVITY MEASUREMENTS

Since the levels of residual radioactivity in these areas were expected to exist at a small fraction of the revised release criteria, with concurrence from the NRC site representative, it was deemed unnecessary to obtain construction material backgrounds for correcting gross beta activity measurements performed on structural and/or system surface SUs. The ambient instrument backgrounds were used in the activity calculations.

Surface activity measurements for beta activity were performed at judgmentally (based on surface scans) selected locations within the SUs to determine if residual activity levels met the release criteria. Forty-eight direct measurements were collected within the SUs where confirmatory surveys were performed (Figures 4 through 14). Direct measurements were collected using GM and hand-held gas proportional detectors coupled to ratemeter-scalers. A smear sample for determining removable gross beta activity levels was collected at each direct measurement location.

MISCELLANEOUS SAMPLING

Nine concrete core samples and four metal roof samples, previously collected by ESL personnel, were submitted to ORISE for radiological analyses. Miscellaneous sampling locations were provided by ESL personnel (Appendix A).

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ORISE's laboratory in Oak Ridge, Tennessee for analysis and interpretation. Sample analyses were performed in accordance with the ORISE Laboratory Procedures Manual (ORISE 2006c). The radionuclides-of-concern (ROC), as identified by ESL, were Sr-90, Co-60 and cesium-137 (Cs-137). Miscellaneous material samples (concrete cores and metal roof) were analyzed by gamma spectroscopy for Co-60 and Cs-137, and Sr-90 by wet chemistry. Gamma spectra were also reviewed for other identifiable total absorption peaks (photopeaks). Miscellaneous material sample results were reported in units of picocuries per gram (pCi/g). Smear samples were analyzed for gross beta activity using a low-background gas proportional counter. Smear results and direct measurements for total surface activity were converted to units of dpm/100 cm^2. Additional information concerning major instrumentation and analytical procedures is provided in Appendices B and C.

FINDINGS AND RESULTS

DOCUMENT REVIEW

ORISE reviewed ESL's DP, FSSP and FSS preliminary data (ESL 2005 and 2006a and b). The procedures, methods, and data submitted by ESL accurately documented the radiological status of the QDP above grade structures per the DP commitments. However, the FSSR for ESL's survey activities in 2006 has not been submitted to ORISE for review.

SURFACE SCANS

The ORISE confirmatory surveys did not detect any elevated radiation levels above the established release criteria within any of the SUs in which surveys were performed. The surface scan results for beta activity indicated several areas that were above background levels. These areas were marked for further investigation.

SURFACE ACTIVITY LEVELS

Direct measurement activity results for the main floor above ground structures ranged from -253 to 48,900 dpm/100 cm^2 for total beta activity. The surface activity level ranges for the SUs surveyed by ORISE are presented in Table 2.

TABLE 2: RANGE OF SURFACE ACTIVITY MEASUREMENTS

Survey Unit	Range of Total Surface Beta Activity (dpm/100 cm^2)	Range of Removable Beta Activity (dpm/100 cm^2)
Mezzanine	-250 to 48,900	-3 to 5
Service Area	20 to 520	-5 to 6
Decon Room	-110 to 16,220	-4 to 3
Chem Lab	130 to 370	-4 to 4
Vestibule	-130 to -40	-3 to 1
Admin Area	-202 to 170	-1 to 4
Reactor Bay	-253 to 310	-2 to 2
Boiler Room	50 to 230	-3 to 4
Area Near Old Dock	-140 to -130	-1 to 1
Hydroblast Area	-10 to 70	-1 to 1
Finishing Area (FA)	-190 to -100	-2 to -1
FA Electrical Room	-80 to 290	-2 to -1
FA Bunker	80	-2
FA Office	-120	1
FA Tool Crib	-168	3
WWTB	-110 to -40	-3 to 1

A complete listing of the confirmatory surface activity results is presented in Table 4.

MISCELLANEOUS SAMPLES

With one exception, the radionuclide concentrations for the concrete and roof samples were at or below the minimum detectable concentration (MDC) for the analytical procedure. The one exception was the roof sample from ESL sampling location #10 (1726M0010) which had a positive value of 0.49 ± 0.12 pCi/g of Cs-137.

COMPARISON OF RESULTS WITH GUIDELINES

The primary ROCs for the QDP are Sr-90 and Co-60 which were identified during characterization as the predominant radionuclides present. The applicable structural and remaining concrete derived concentration guideline levels (DCGLs) specified in the DP and approved by the NRC are as follows (ESL 2006b):

TABLE 3: DERIVED CONCENTRATION GUIDELINE LEVELS FROM QUEHANNA DECOMMISSIONING PROJECT DP

Media	DCGL[a]	Note
Above grade structures	250,000 dpm/100 cm^2 for Sr-90 total surface contamination	Removable contamination will be controlled to Reg. Guide 1.86 levels of 200 dpm/100 cm^2
Remaining concrete	30,000 pCi/g	Concrete includes any remaining cinder blocks that will be used as fill

[a]DCGL values taken from the LTP and LTR (ESL 2006b).

All direct measurement, smear, and miscellaneous sample results, presented in Tables 4 and 5, were less than the applicable DCGLs as listed in Table 3.

SUMMARY

At the request of the Office of Federal and State Materials and Environmental Management Programs, U.S. Nuclear Regulatory Commission (NRC), the Oak Ridge Institute for Science and Education (ORISE) conducted confirmatory surveys of the Quehanna Decommissioning Project (QDP) above grade structures during the period of December 5 and 6, 2006. The survey activities consisted of visual inspections and radiological surveys including beta and gamma surface scans and surface beta activity measurements. Cursory beta and gamma scans were performed on below grade structures in the basement. ORISE did not perform surface activity measurements in the basement areas since previous ORISE confirmatory surveys did not identify residual surface contamination in those areas. ORISE also performed radiological analyses on 13 concrete and metal roof samples that were previously collected by Energy*Solutions*, LLC (ESL) personnel.

The results of the confirmatory surveys indicated that the beta surface activity levels were less than the applicable NRC-approved release criteria for the QDP. All confirmatory surface activity level results were less than the derived concentration guideline levels (DCGLs) for the Sr-90 as specified in the decommissioning plan [DP (ESL 2006a)]. The ORISE results are also consistent with the radiological survey results in the final status survey (FSS) preliminary data provided to ORISE for review.

FIGURES

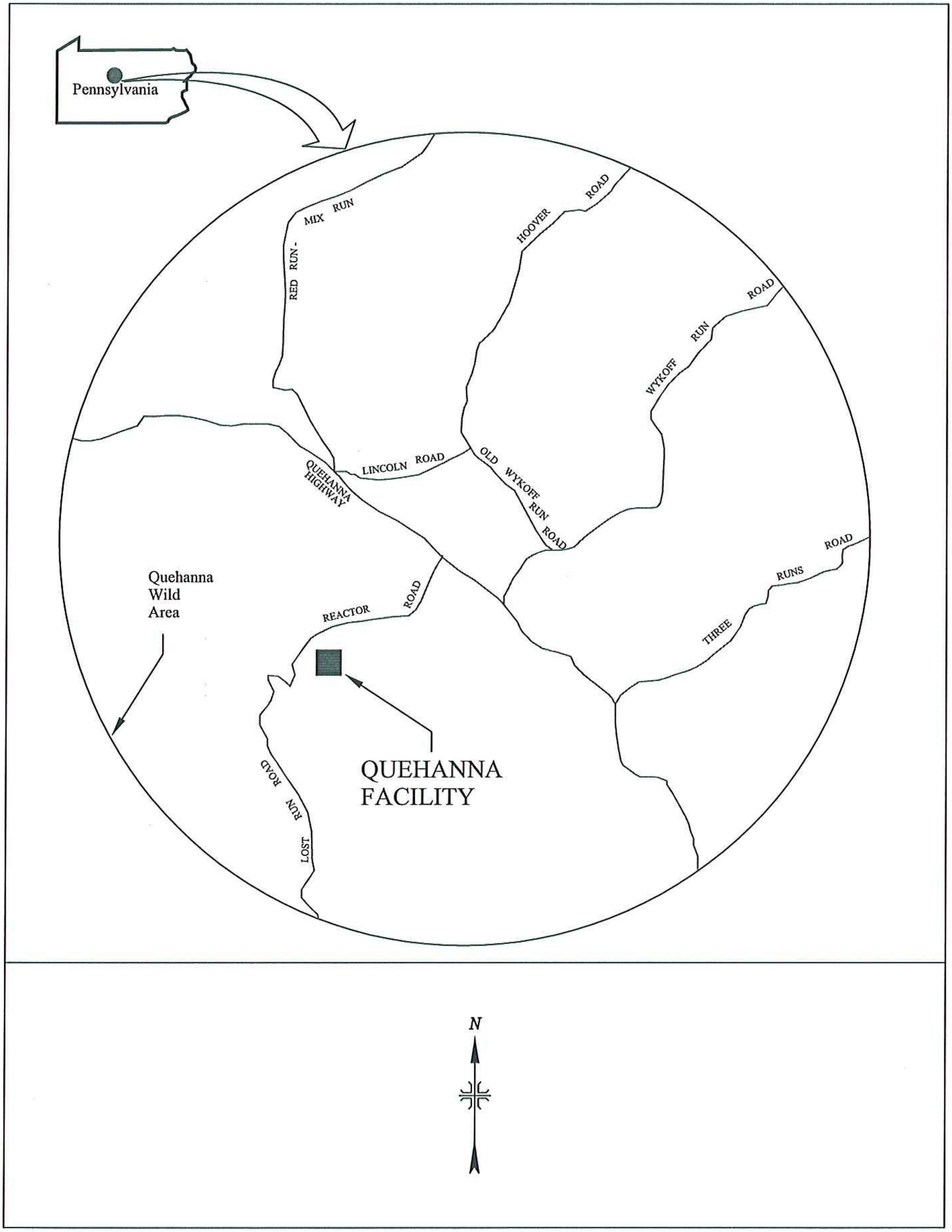


FIGURE 1: Location of the Quehanna Facility - Karthaus, Pennsylvania

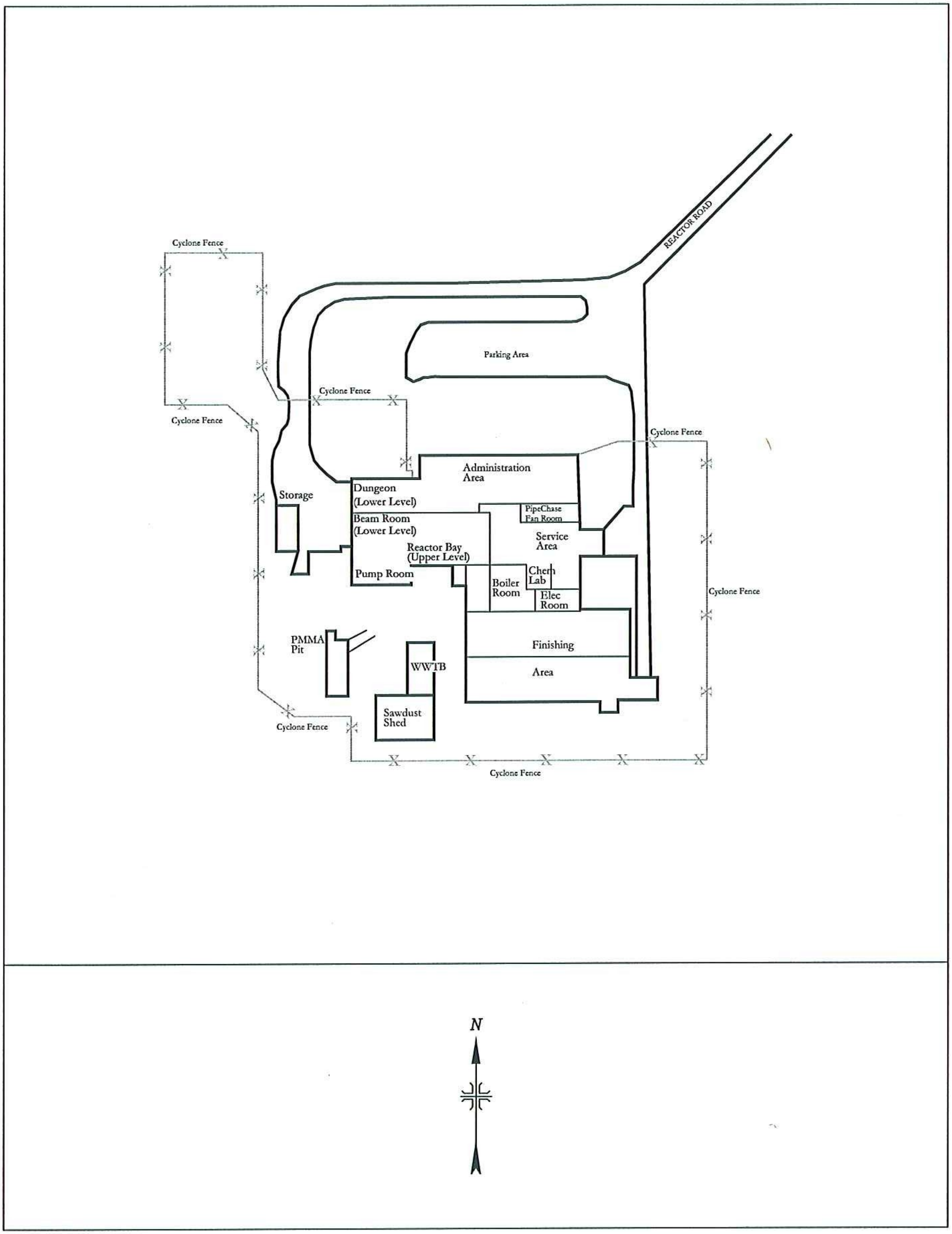


FIGURE 2: Plot Plan of the Quehanna Facility - Karthaus, Pennsylvania

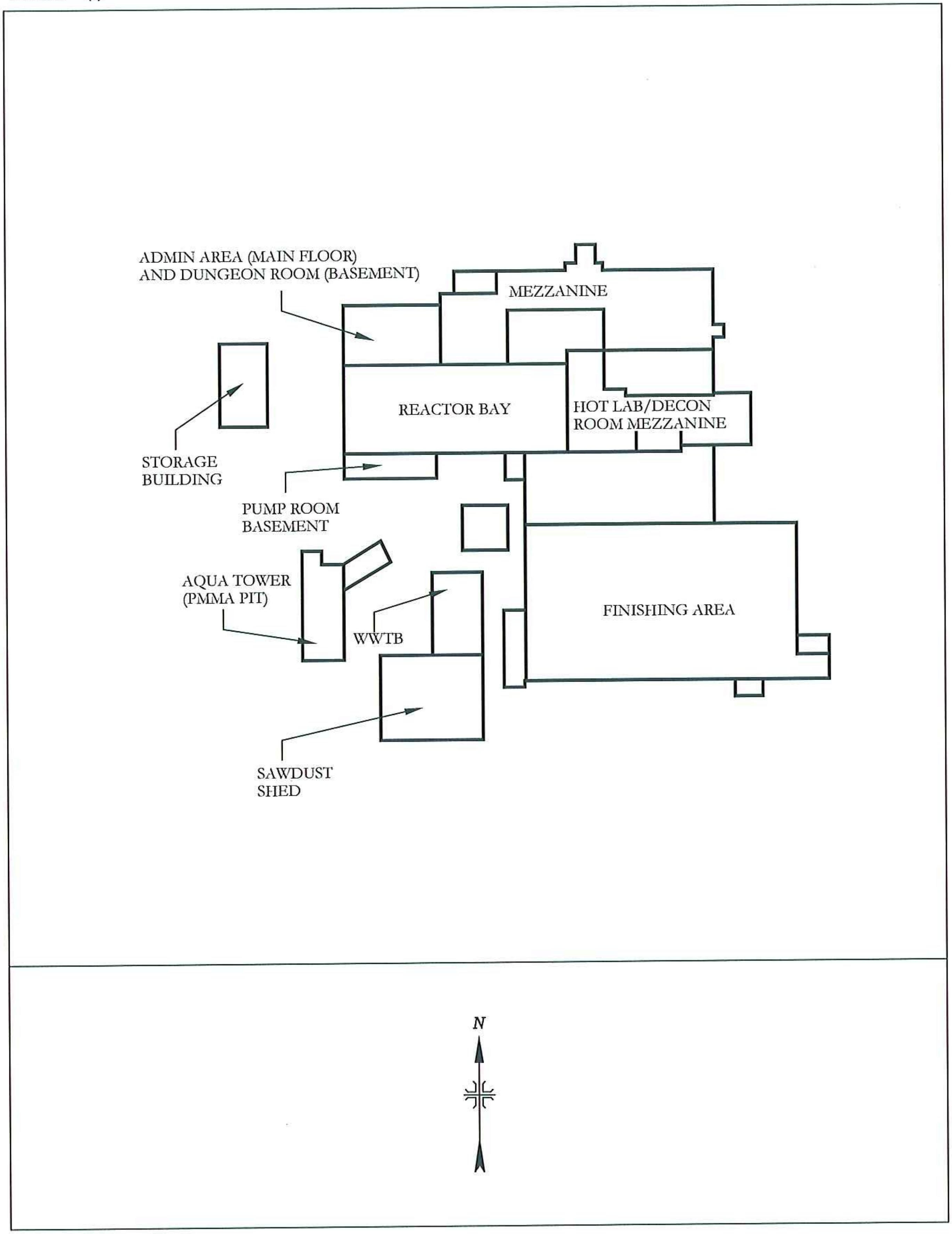


FIGURE 3: Quehanna Facility Floor Plan

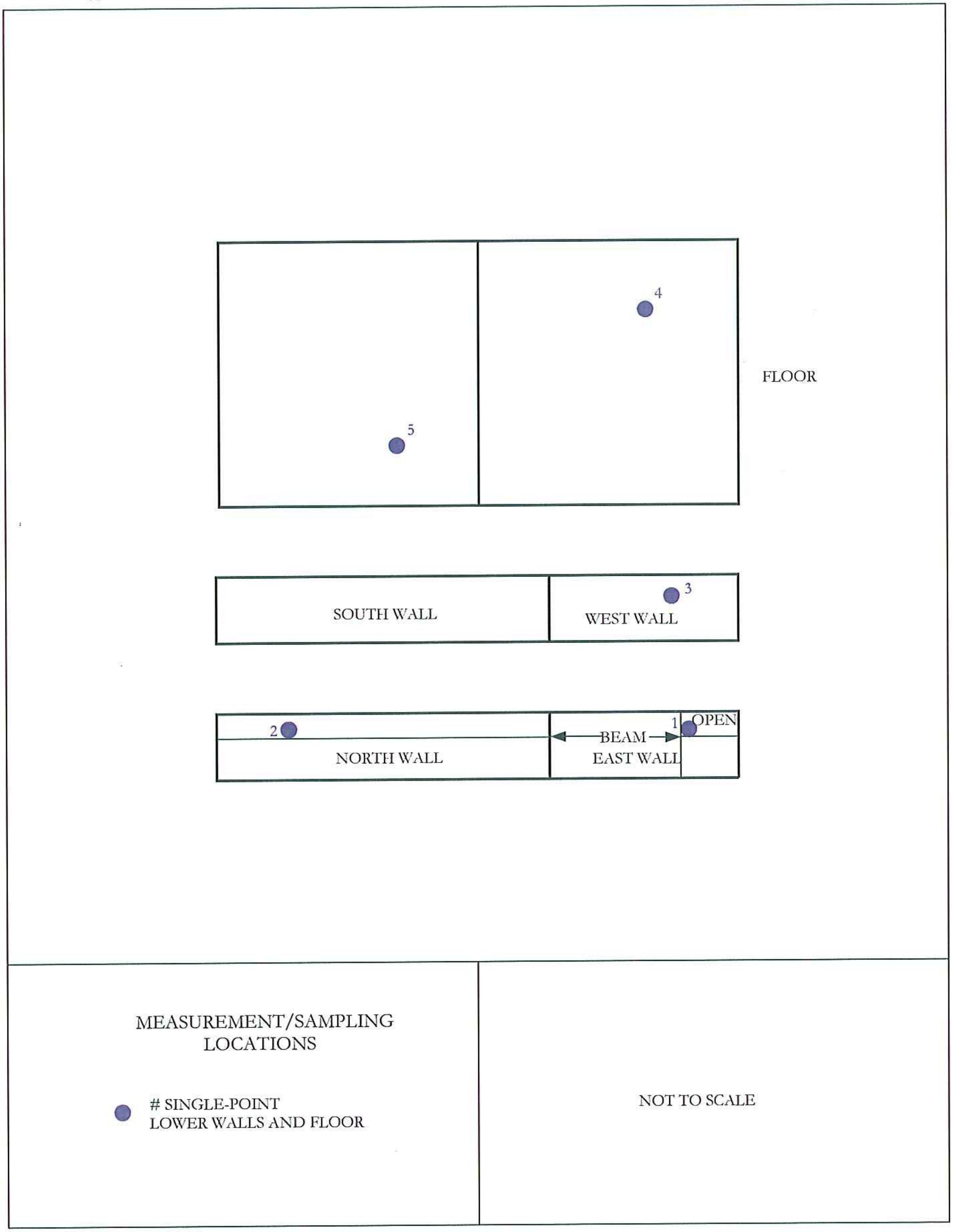


FIGURE 4: Office Mezzanine Area - Measurement and Sampling Locations

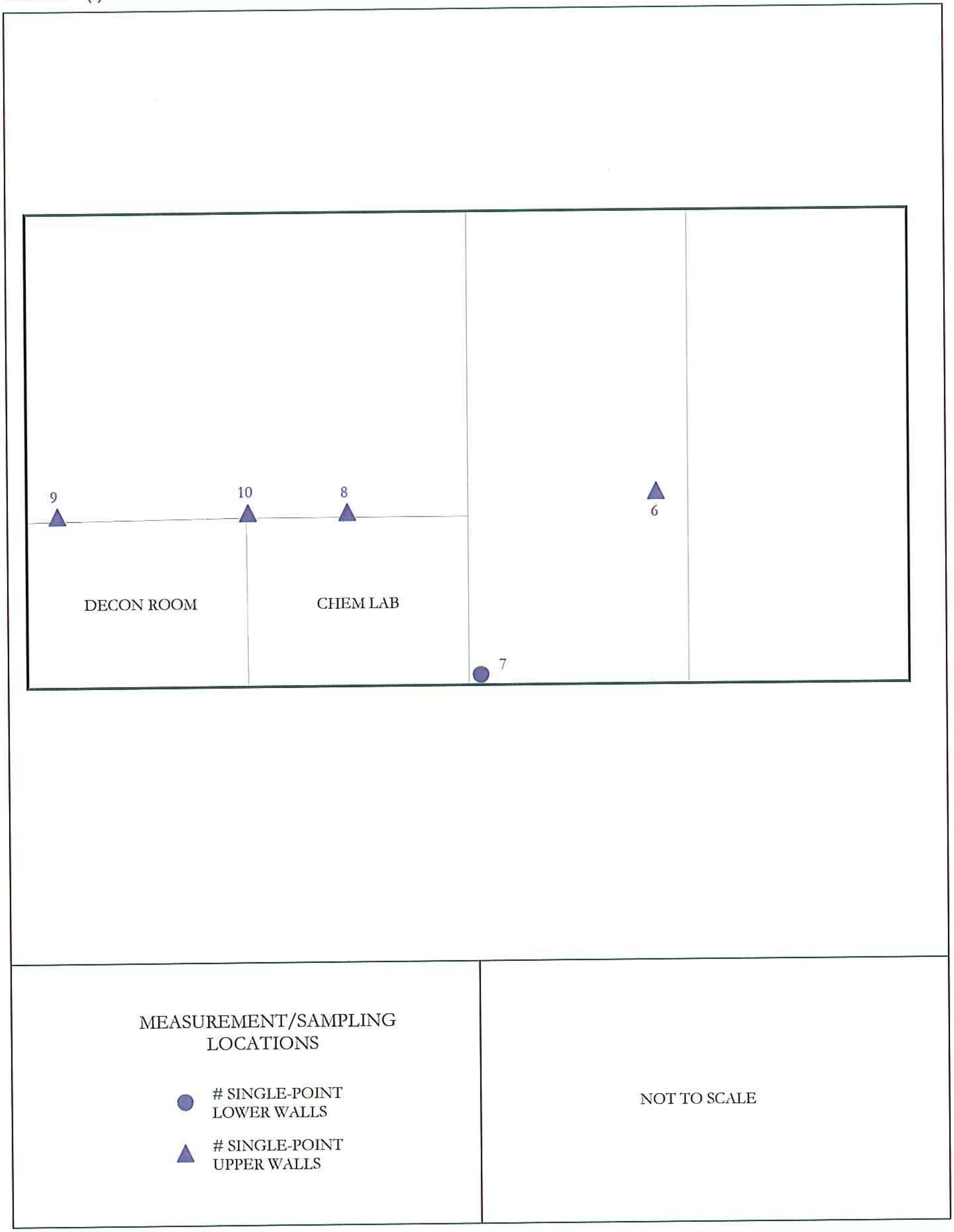


FIGURE 5: Service Area, South Wall - Measurement and Sampling Locations

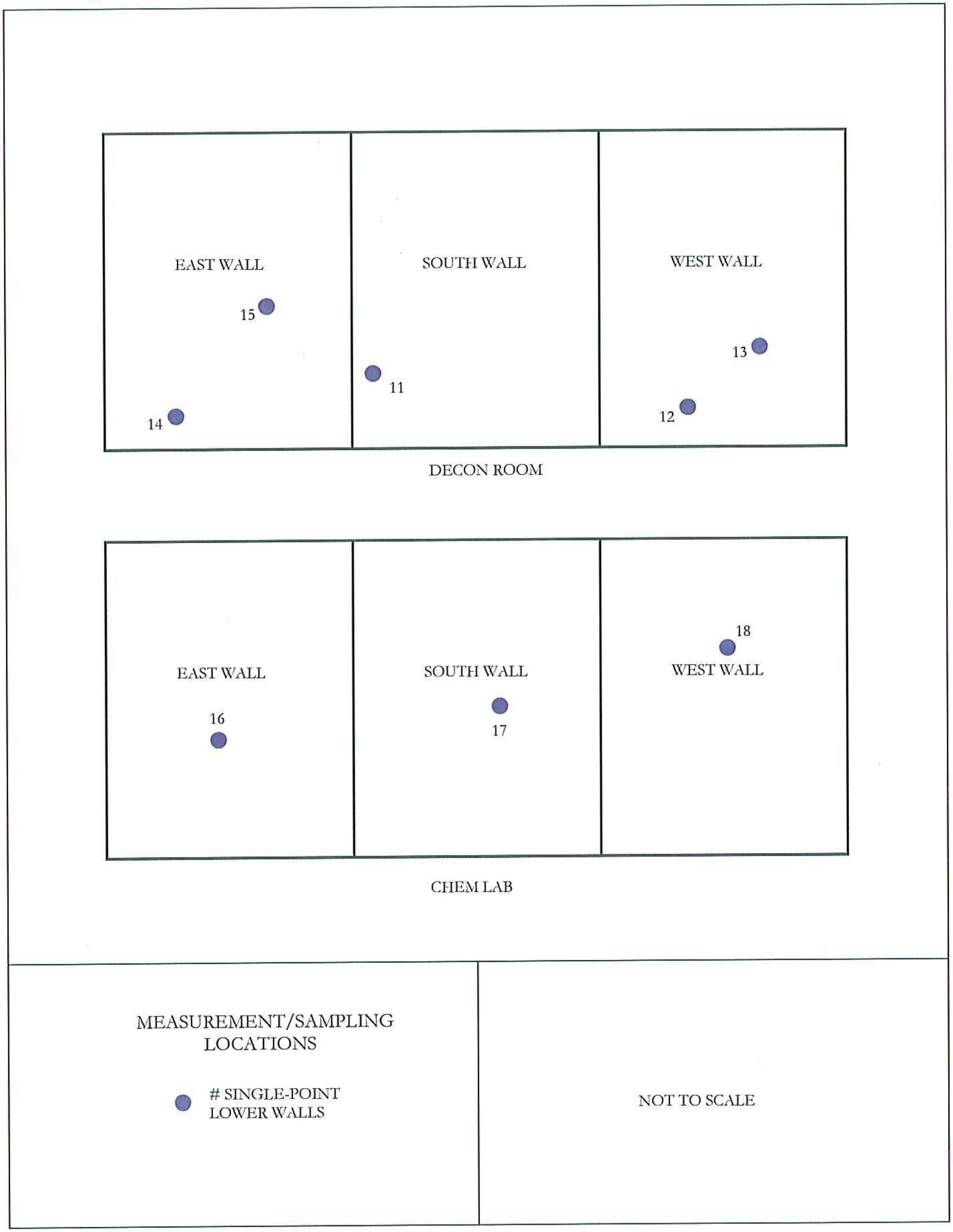


FIGURE 6: Chem Lab and Decon Room Walls - Measurement and Sampling Locations

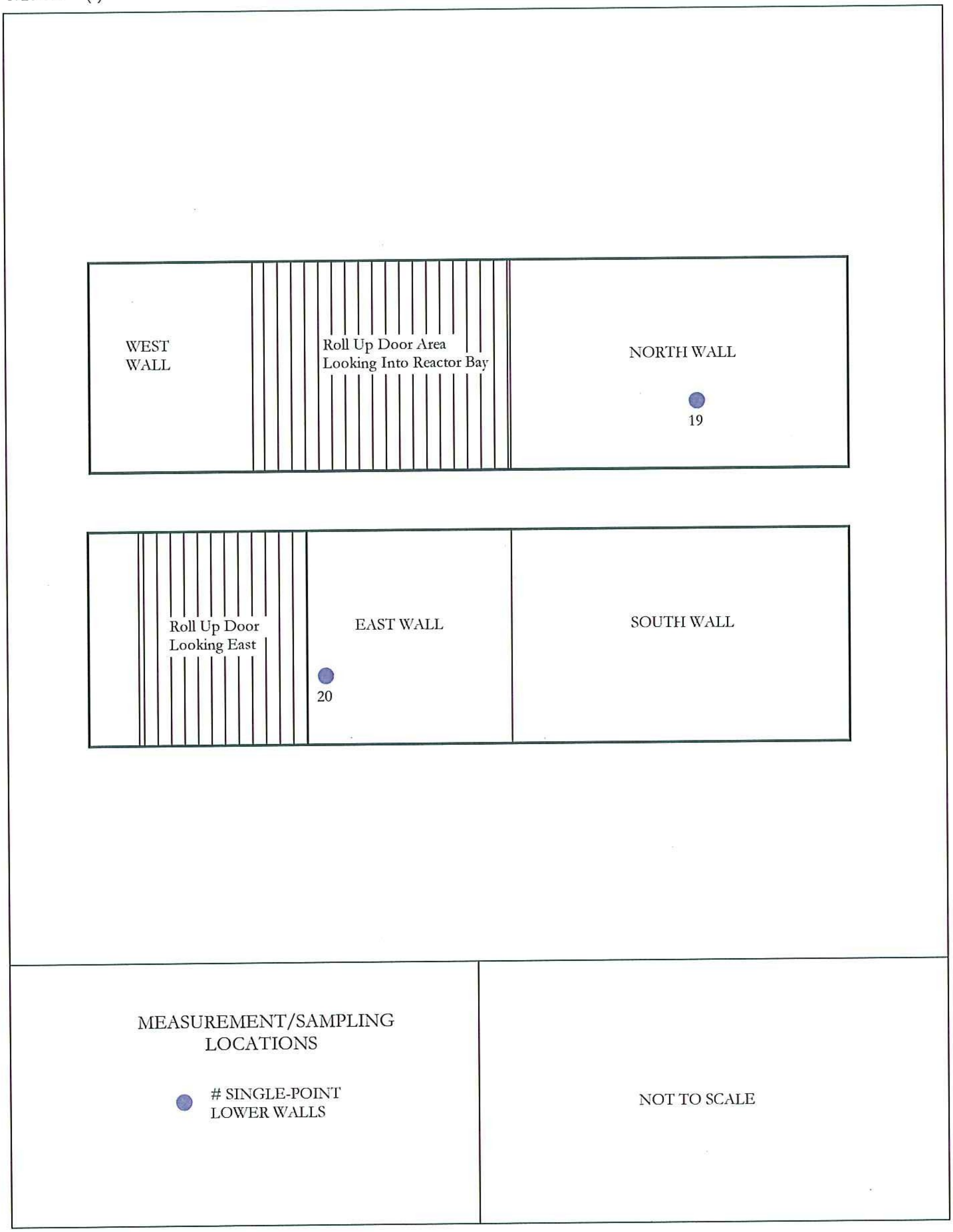


FIGURE 7: Service Area Vestibule - Measurement and Sampling Locations

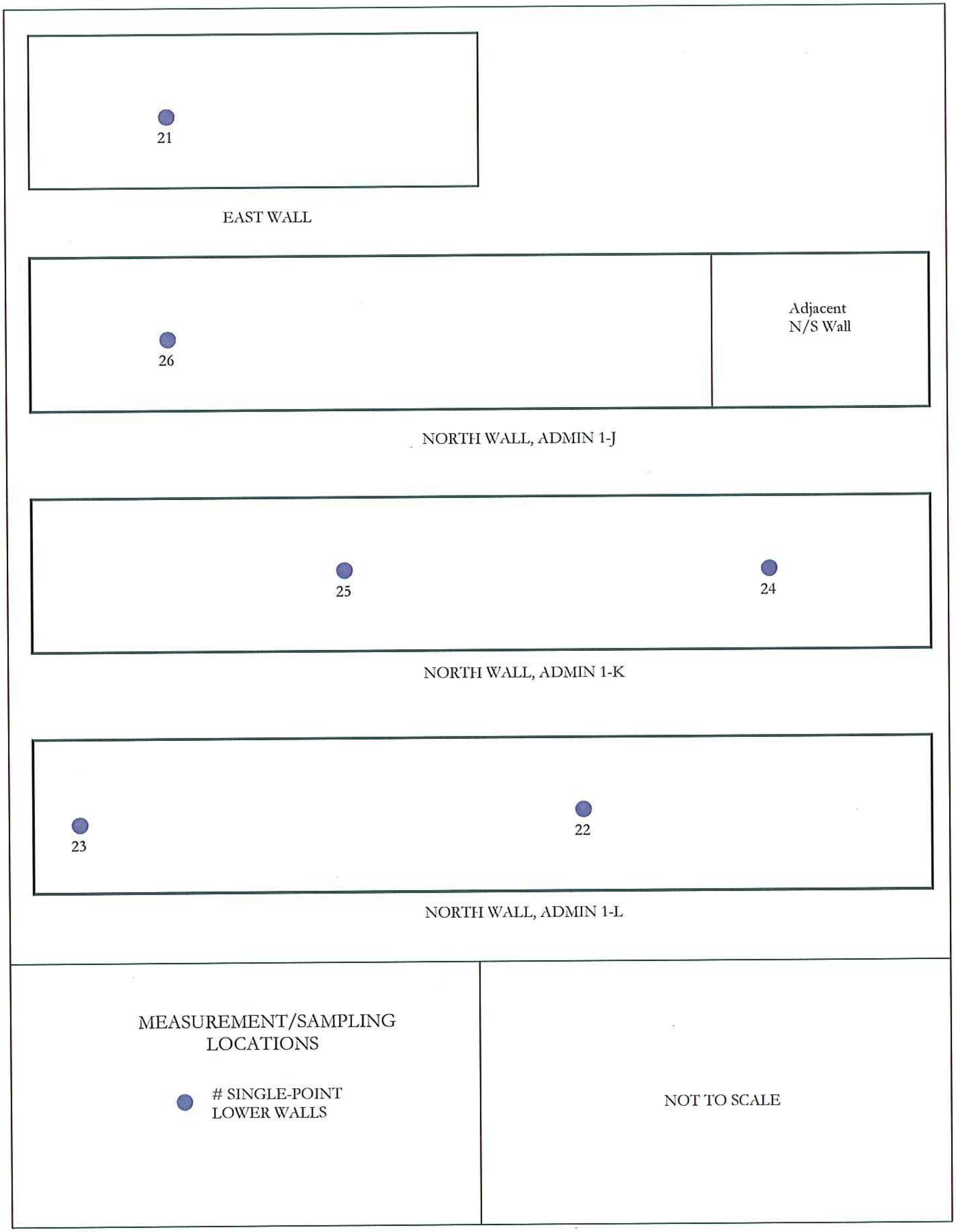


FIGURE 8: Admin Area, North and East Walls - Measurement and Sampling Locations

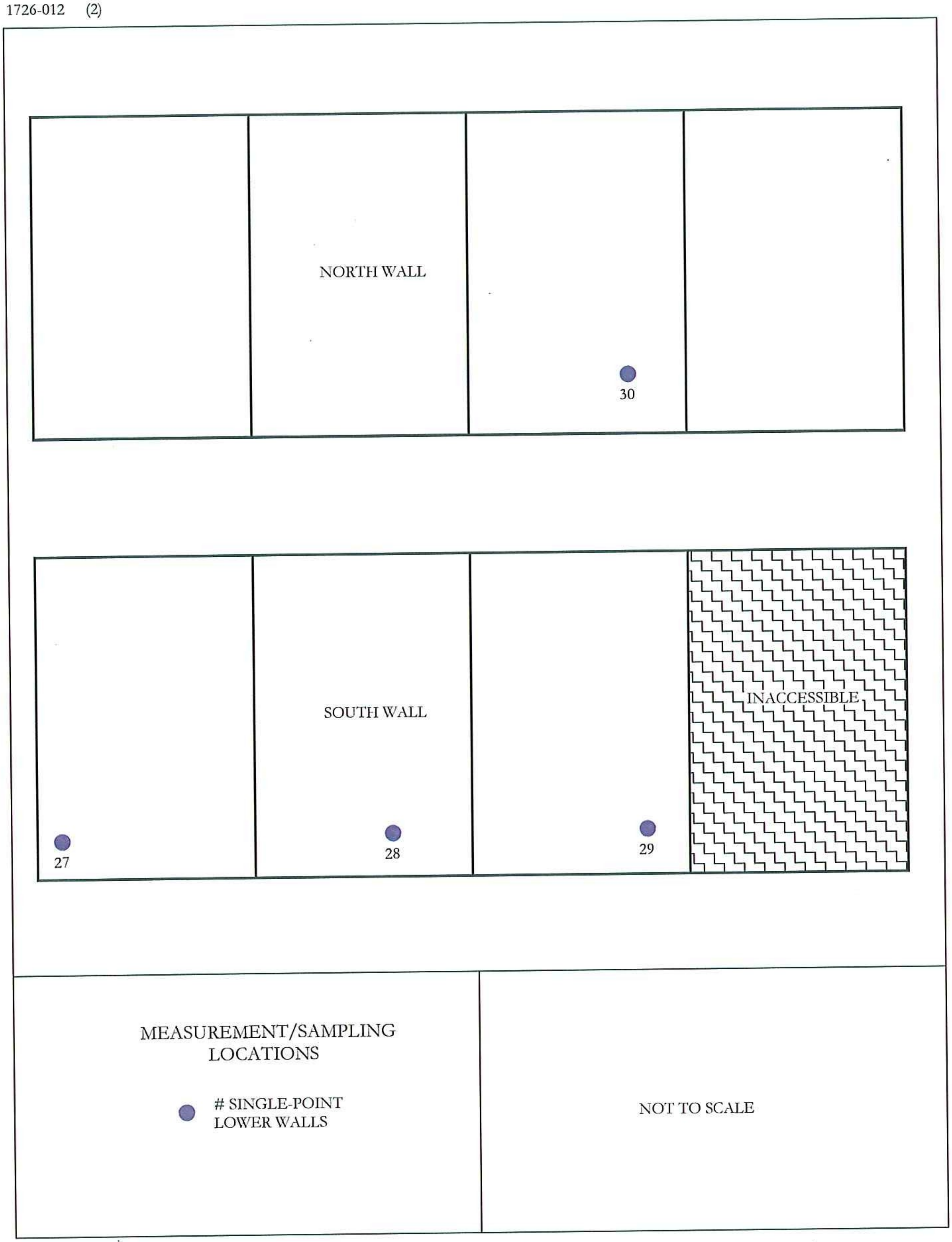


FIGURE 9: Reactor Bay, North and South Walls - Measurement and Sampling Locations

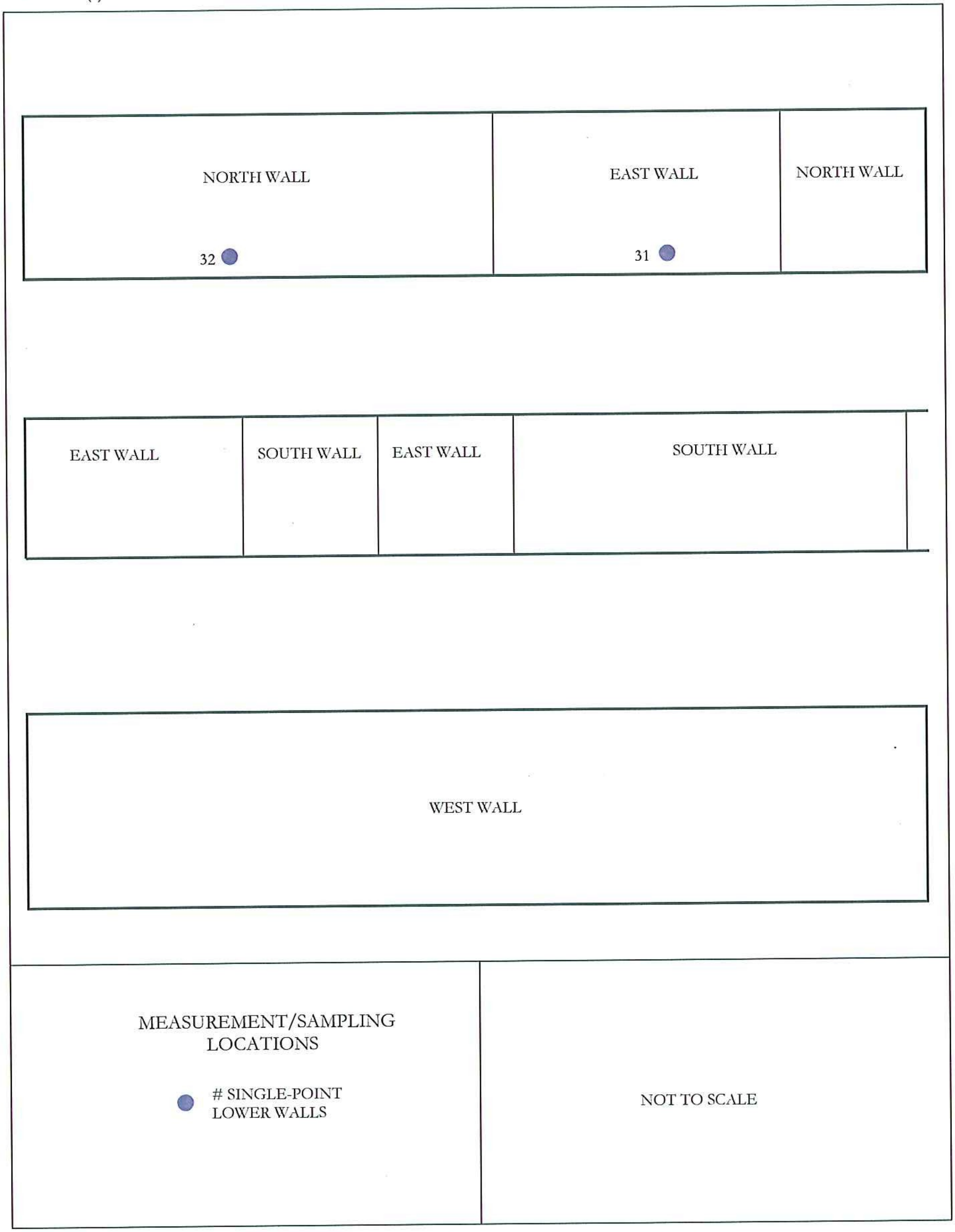


FIGURE 10: Boiler Room Walls - Measurement and Sampling Locations

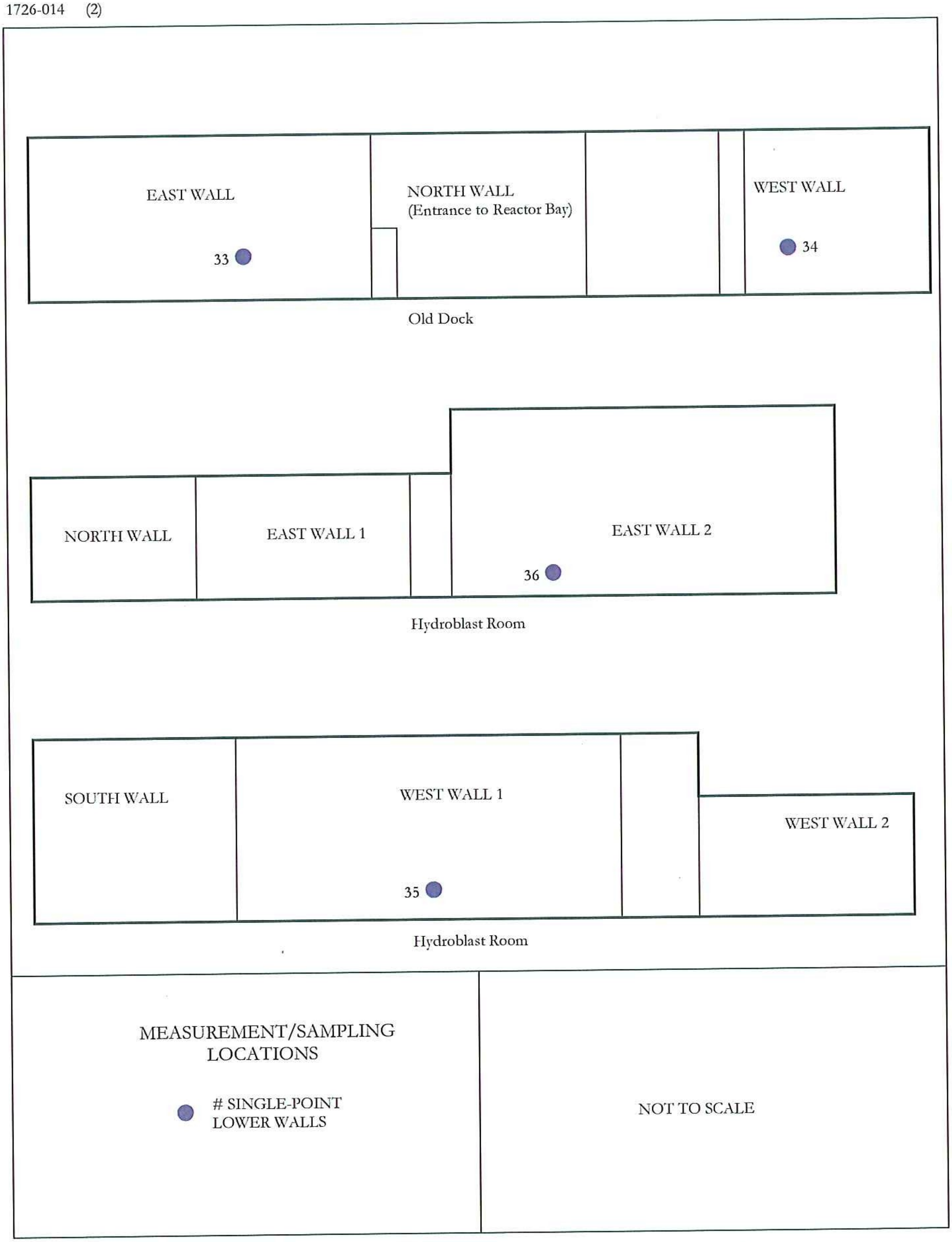


FIGURE 11: Old Dock and Hydroblast Room - Measurement and Sampling Locations

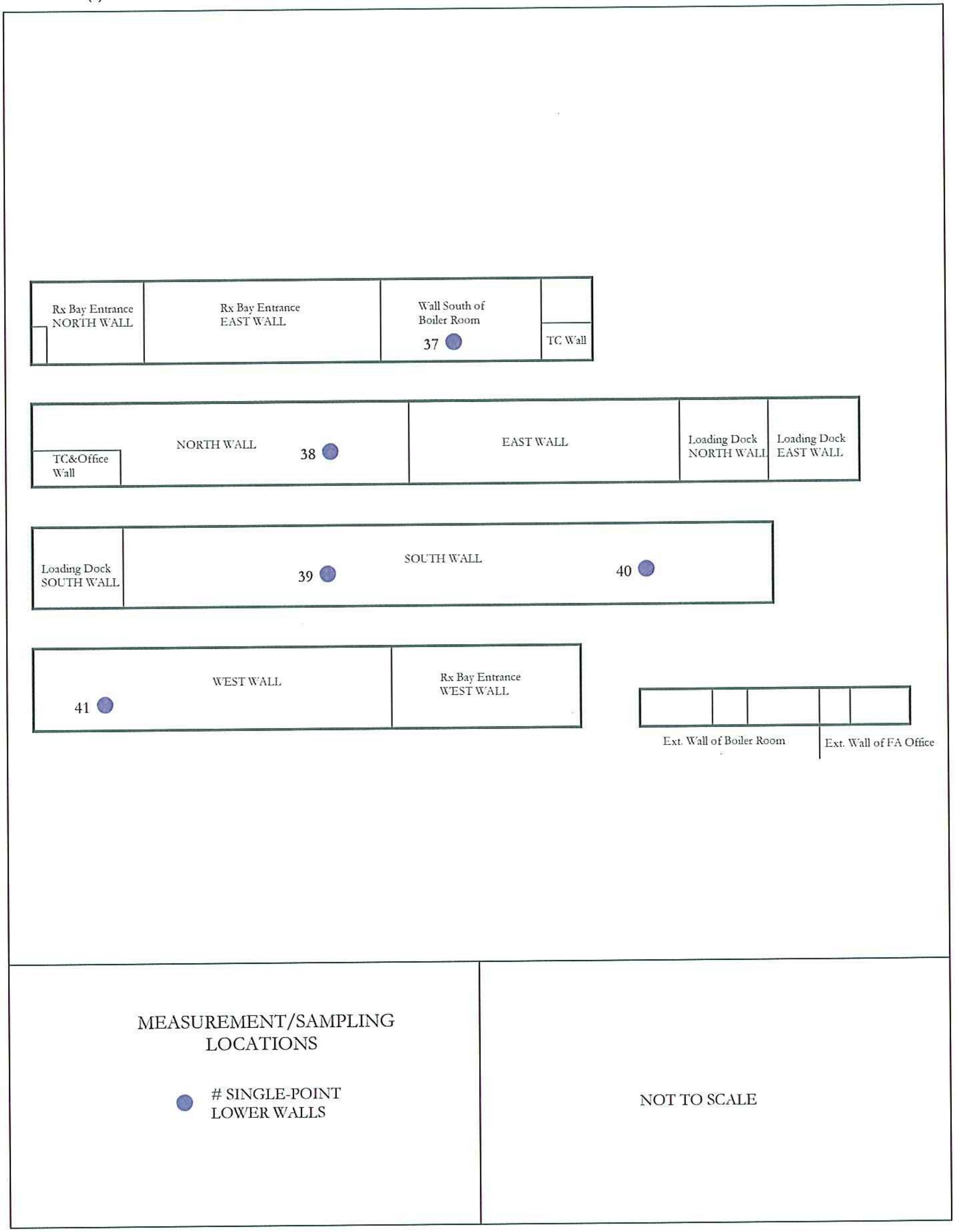


FIGURE 12: Finishing Area Walls - Measurement and Sampling Locations

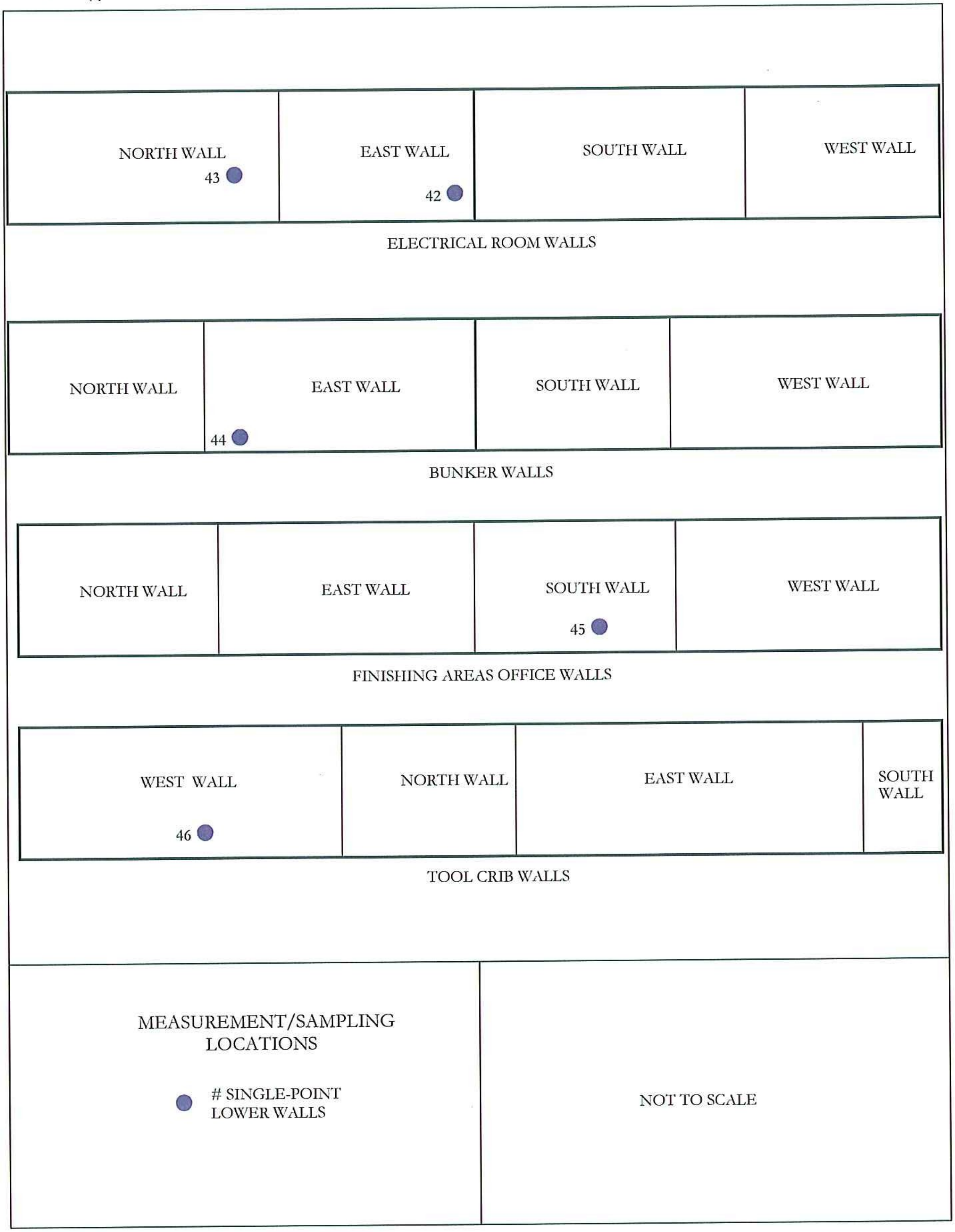


FIGURE 13: Finishing Area Office, Electrical Room, Bunker Office and Tool Crib Walls - Measurement and Sampling Locations

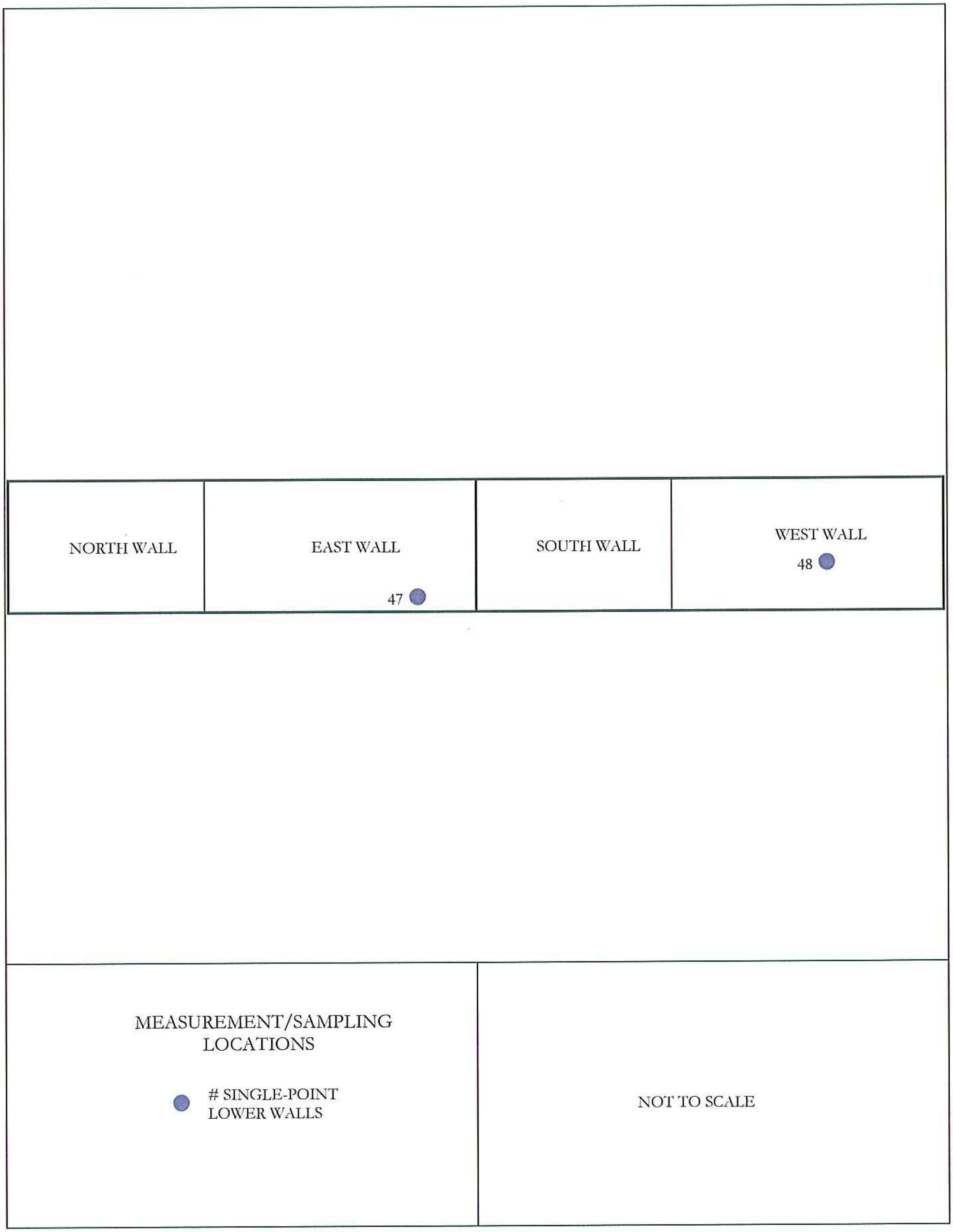


FIGURE 14: Waste Water Treatment Building - Measurement and Sampling Locations

TABLES

TABLES

TABLE 4

SURFACE ACTIVITY MEASUREMENTS
QUEHANNA DECOMMISSIONING PROJECT
KARTHAUS, PENNSYLVANIA

Measurement Location[a]	Surface Type	Beta Surface Activity (dpm/100 cm^2)	Removable Beta Activity (dpm/100 cm^2)
Mezzanine			
1 LW	Metal	48,900 ± 1,800[b]	3 ± 6
2 LW	Metal	-250 ± 320	-1 ± 5
3 LW	Metal	-170 ± 330	5 ± 7
4 FL	Concrete	-170 ± 330	-3 ± 4
5 FL	Concrete	-140 ± 330	1 ± 5
Service Area			
6 UW	Metal	240 ± 120	3 ± 6
7 LW	Concrete	350 ± 120	-5 ± 3
8 UW	Concrete	520 ± 130	2 ± 6
9 UW	Concrete	20 ± 110	6 ± 7
10 UW	Concrete	180 ± 120	3 ± 6
Decon Room			
11 LW	Concrete	15,130 ± 400	3 ± 6
12 LW	Concrete	16,220 ± 410	-4 ± 3
13 LW	Concrete	4,850 ± 240	-4 ± 3
14 LW	Concrete	8,730 ± 310	-3 ± 4
15 LW	Metal	-110 ± 100	-2 ± 5
Chem Lab			
16 LW	Concrete	130 ± 110	4 ± 7
17 LW	Concrete	270 ± 120	-4 ± 3
18 LW	Concrete	370 ± 120	-1 ± 5
Vestibule			
19 LW	Metal	-40 ± 110	-3 ± 4
20 LW	Metal	-130 ± 100	1 ± 5

TABLE 4 (continued)

SURFACE ACTIVITY MEASUREMENTS
QUEHANNA DECOMMISSIONING PROJECT
KARTHAUS, PENNSYLVANIA

Measurement Location[a]	Surface Type	Beta Surface Activity (dpm/100 cm^2)	Removable Beta Activity (dpm/100 cm^2)
Admin Area			
21 LW	Metal	170 ± 120	1 ± 5
22 LW	Metal	-120 ± 100	4 ± 7
23 LW	Metal	-30 ± 110	-1 ± 5
24 LW	Metal	-80 ± 100	1 ± 5
25 LW	Metal	60 ± 110	4 ± 7
26 LW	Metal	-202 ± 98	-1 ± 5
Reactor Bay			
27 LW	Metal	70 ± 110	2 ± 6
28 LW	Metal	-253 ± 95	-2 ± 5
29 LW	Metal	310 ± 120	-1 ± 5
30 LW	Metal	-202 ± 98	-1 ± 5
Boiler Room			
31 LW	Concrete	230 ± 120	-3 ± 4
32 LW	Concrete	50 ± 110	4 ± 7
Area Near Old Dock			
33 LW	Concrete	-140 ± 100	1 ± 5
34 LW	Metal	-130 ± 100	-1 ± 5
Hydroblast Area			
35 LW	Concrete	-10 ± 110	-1 ± 5
36 LW	Concrete	70 ± 110	1 ± 5
Finishing Area (FA)			
37 LW	Concrete	-120 ± 100	-1 ± 5
38 LW	Metal	-173 ± 99	-2 ± 5
39 LW	Metal	-188 ± 98	-2 ± 5
40 LW	Metal	-190 ± 98	-2 ± 5
41 LW	Metal	-100 ± 100	-2 ± 5

TABLE 4 (continued)

SURFACE ACTIVITY MEASUREMENTS
QUEHANNA DECOMMISSIONING PROJECT
KARTHAUS, PENNSYLVANIA

Measurement Location[a]	Surface Type	Beta Surface Activity (dpm/100 cm^2)	Removable Beta Activity (dpm/100 cm^2)
Electrical Room			
42 LW	Concrete	-80 ± 100	-2 ± 5
43 LW	Concrete	290 ± 120	-1 ± 5
Bunker			
44 LW	Concrete	80 ± 110	-2 ± 5
Office			
45 LW	Concrete	-120 ± 100	1 ± 5
Tool Crib			
46 LW	Concrete	-168 ± 99	3 ± 6
Waste Water Treatment Building			
47 LW	Metal	-110 ± 100	1 ± 5
48 LW	Metal	-40 ± 110	-3 ± 4

[a]Refer to Figures 4 to 14. FL = floor, LW = lower wall, and UW = upper wall.
[b]Uncertainties represent the 95% confidence level based on counting statistics only.

TABLE 5

RADIONUCLIDE CONCENTRATIONS IN MISCELLANEOUS SAMPLES
QUEHANNA DECOMMISSIONING PROJECT
KARTHAUS, PENNSYLVANIA

Sample Identification		Sample Type	Radionuclide Concentrations (pCi/g)		
ORISE	**Energy*Solutions*[a]**		**Co-60[b]**	**Sr-90[c]**	**Cs-137[b]**
1726M0001	25	Concrete	0.01 ± 0.06[d] (0.11)[e]	0.34 ± 0.25 (0.40)	0.04 ± 0.05 (0.09)
1726M0002	26	Concrete	0.01 ± 0.05 (0.09)	0.35 ± 0.28 (0.46)	0.08 ± 0.07 (0.07)
1726M0003	42	Concrete	-0.01 ± 0.04 (0.08)	0.37 ± 0.27 (0.43)	-0.03 ± 0.05 (0.06)
1726M0004	51	Concrete	0.02 ± 0.04 (0.08)	0.23 ± 0.26 (0.43)	0.03 ± 0.04 (0.07)
1726M0005	61	Concrete	-0.03 ± 0.05 (0.08)	0.27 ± 0.23 (0.38)	0.05 ± 0.04 (0.07)
1726M0006	64	Concrete	0.01 ± 0.04 (0.08)	0.19 ± 0.26 (0.44)	0.01 ± 0.04 (0.07)
1726M0007	72	Concrete	0.00[f] ± 0.03 (0.06)	0.06 ± 0.25 (0.44)	0.00 ± 0.04 (0.06)
1726M0008	89	Concrete	0.01 ± 0.04 (0.08)	0.03 ± 0.24 (0.43)	-0.02 ± 0.05 (0.08)
1726M0009	92	Concrete	-0.02 ± 0.06 (0.10)	0.31 ± 0.29 (0.48)	-0.01 ± 0.05 (0.08)
1726M0010	6	Roof	0.00 ± 0.06 (0.11)	0.28 ± 0.31 (0.51)	0.49 ± 0.12 (0.10)
1726M0011[g]	13	Roof	0.00 ± 0.02 (0.04)	-0.06 ± 0.49 (0.89)	0.08 ± 0.03 (0.03)
1726M0012[g]	14	Roof	0.06 ± 0.22 (0.34)	0.31 ± 0.40 (0.69)	0.34 ± 0.26 (0.23)
1726M0013[g]	17	Roof	3.6 ± 7.9 (14)	0.09 ± 0.43 (0.77)	9 ± 10 (14)

[a]Sample identifications provided by Energy*Solutions*.
[b]Analysis by gamma spectroscopy.
[c]Analysis by wet chemistry.
[d]Uncertainties represent the 95% confidence level base on total propagated uncertainties.
[e]Minimum detectable concentrations (MDC) for the analytical results are in parentheses.
[f]Zero values due to rounding.
[g]Co-60 and Cs-137 analytical results for these samples are qualified due to gamma spectroscopy geometry problems associated with the sample.

REFERENCES

Energy*Solutions*, LLC (ESL). Decommissioning Plan for the Quehanna, Pennsylvania Site. Document Number 82A9089, Revision 3. New Milford, Connecticut; January 2005.

Energy*Solutions*, LLC. Quehanna Decommissioning Project Final Status Survey Report. Document Number 82A9554, Revision 0. New Milford, Connecticut; January 2005.

Energy*Solutions*, LLC. Decommissioning Plan for the Quehanna, Pennsylvania Site. Document number 82A9089, Revision 4. New Milford, Connecticut; March 2006a.

Energy*Solutions*, LLC. Final Status Survey Plan for the Quehanna, Pennsylvania Site. Document Number 82A9604, Revision 1. New Milford, Connecticut; October 2006b.

Oak Ridge Institute for Science and Education (ORISE). Interim Confirmatory Letter Report for the Quehanna Decommissioning Project, Karthaus, Pennsylvania [Docket No. 30-29288; RFTA No. 04-008]. Oak Ridge, Tennessee; February 11, 2005a.

Oak Ridge Institute for Science and Education. Confirmatory Survey Results for Class 1 and Class 2 Interior Building Areas, Quehanna Decommissioning Project, Karthaus, Pennsylvania [Docket No. 30-29288; RFTA no. 04-008]. Oak Ridge, Tennessee; May 24, 2005b.

Oak Ridge Institute for Science and Education. Final Confirmatory Survey Plan for the Quehanna Decommissioning Project in Karthaus, Pennsylvania (Docket No. 30-29288; RFTA No. 06-012). Oak Ridge, Tennessee; November 30, 2006a.

Oak Ridge Institute for Science and Education. Survey Procedures Manual for the Independent Environmental Assessment and Verification Program. Oak Ridge, Tennessee; August 7, 2006b.

Oak Ridge Institute for Science and Education. Laboratory Procedures Manual for the Environmental Survey and Site Assessment Program. Oak Ridge, Tennessee; April 18, 2006c.

Oak Ridge Institute for Science and Education. Quality Program Manual for the Independent Environmental Assessment and Verification Program. Oak Ridge, Tennessee; March 1, 2007.

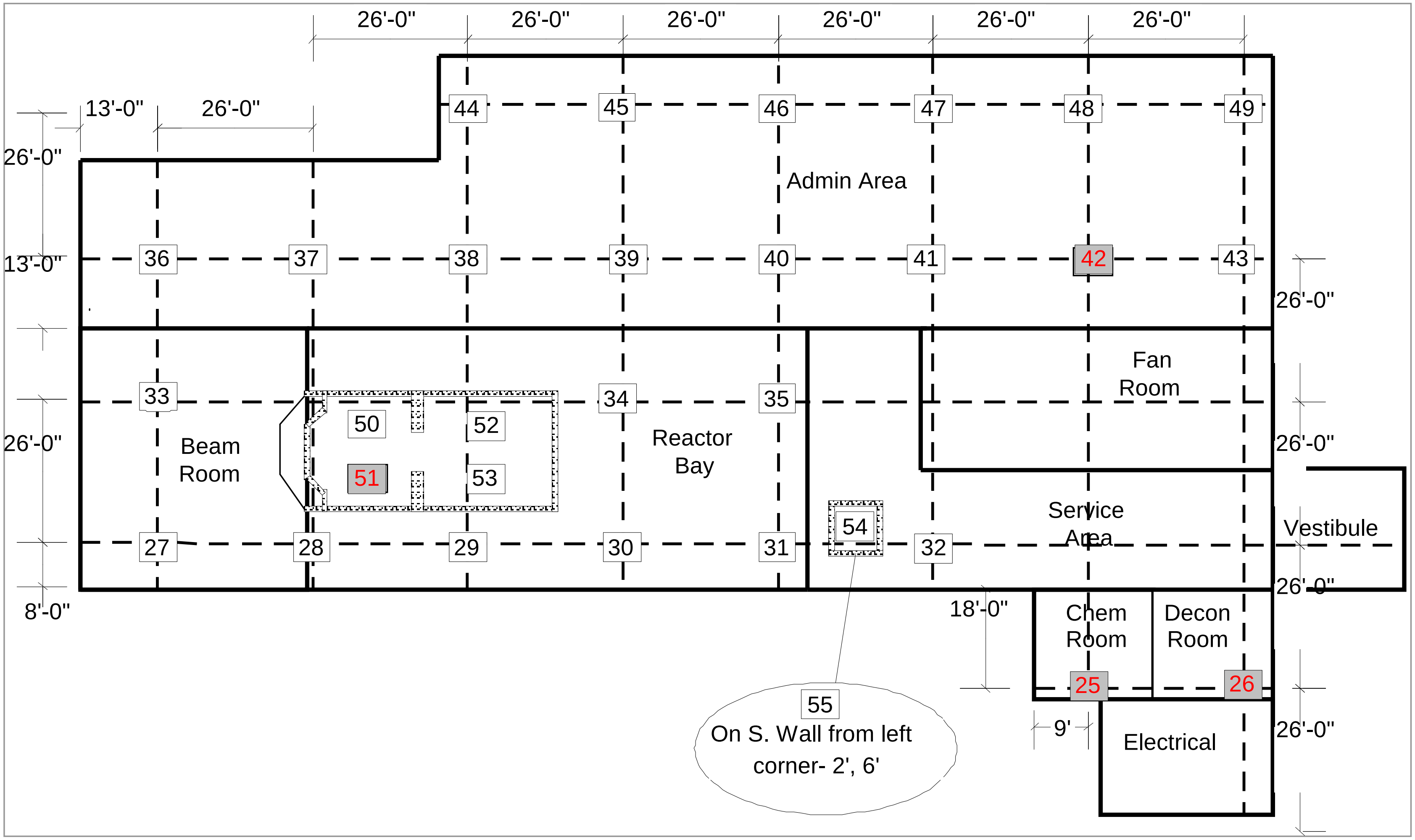
Scientech, Incorporated (STI). Quehanna Decommissioning Project, Final Status Survey Plan. Revision 0, New Milford, Connecticut; June 22, 2004.

U.S. Atomic Energy Commission (AEC). Regulatory Guide 1.86 – Termination of Operating Licenses for Nuclear Reactors. Washington, DC; June 1974.

U.S. Nuclear Regulatory Commission (NRC). Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). NUREG-1575; Revision 1. Washington, DC; August 2000.

APPENDIX A

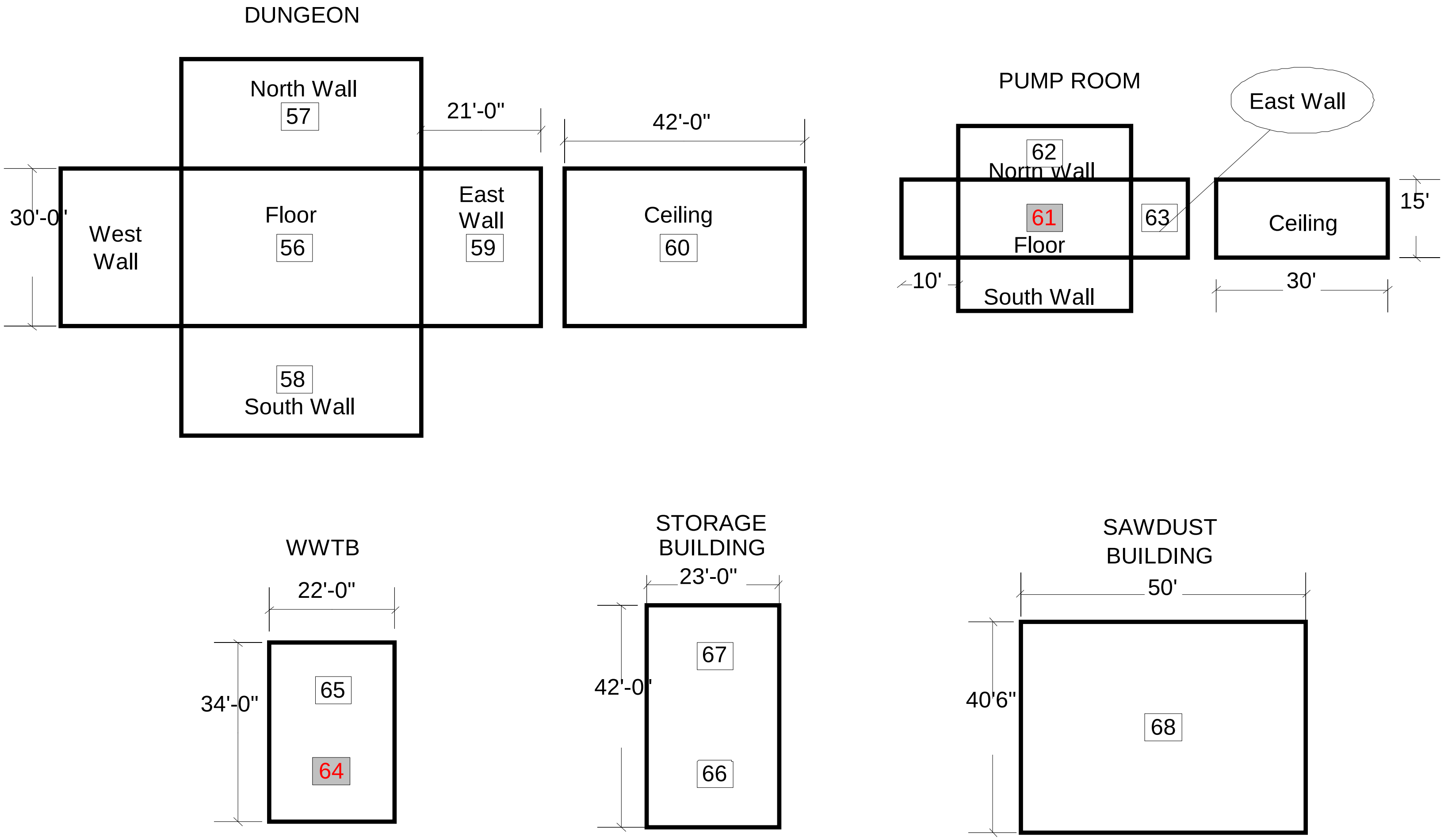
MISCELLANEOUS SAMPLE LOCATIONS



SAMPLING LOCATIONS

#	Energy*Solutions* , LLC Sampling Locations
#	Samples Selected by ORISE for Radiological Analyses

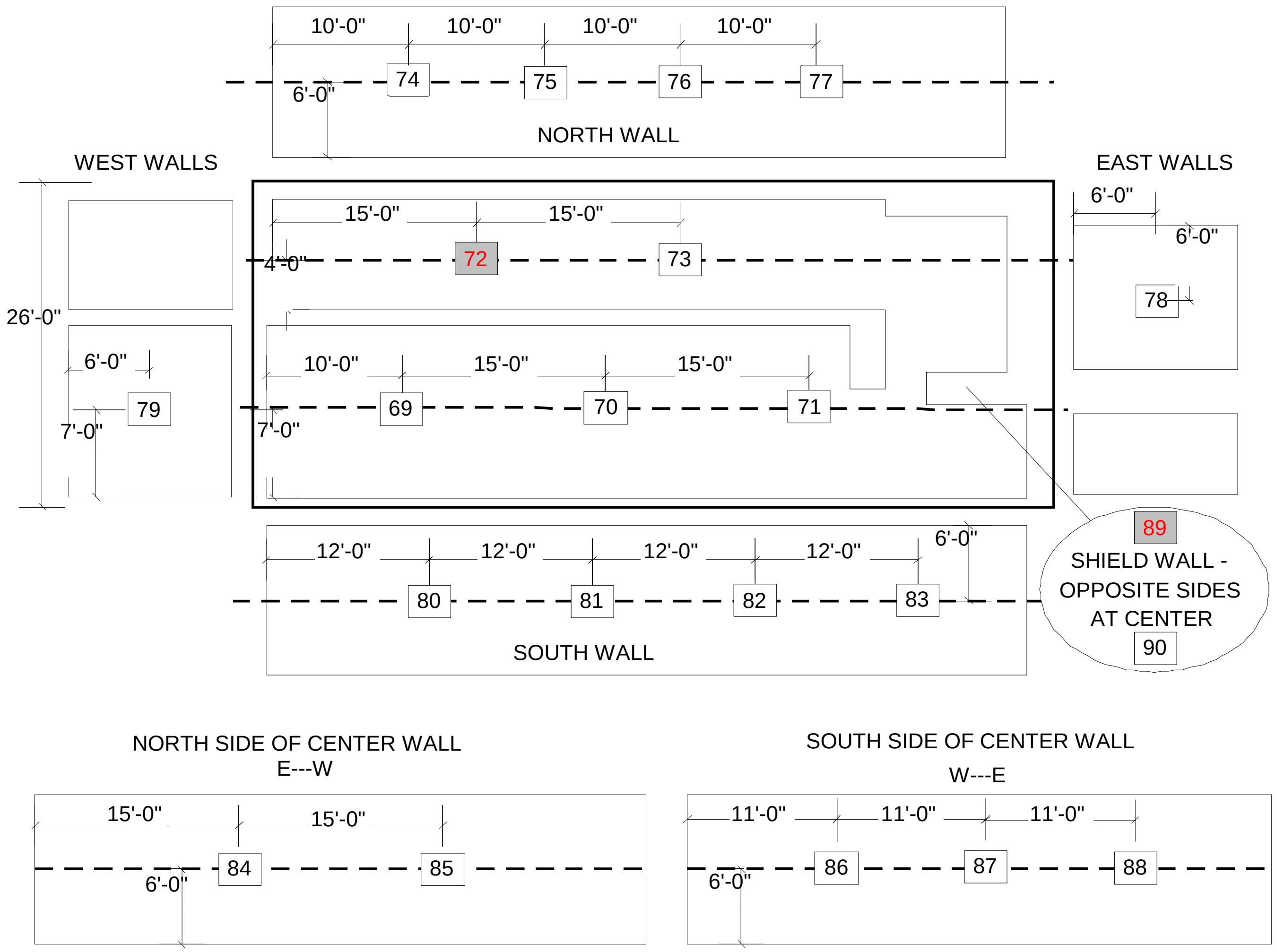
Sample Figure 1: Energy*Solutions* , LLC— Sampling Locations



SAMPLING LOCATIONS

#	Energy*Solutions*, LLC Sampling Locations
#	Samples Selected by ORISE for Radiological Analyses

Sample Figure 2: Energy*Solutions*, LLC— Sampling Locations

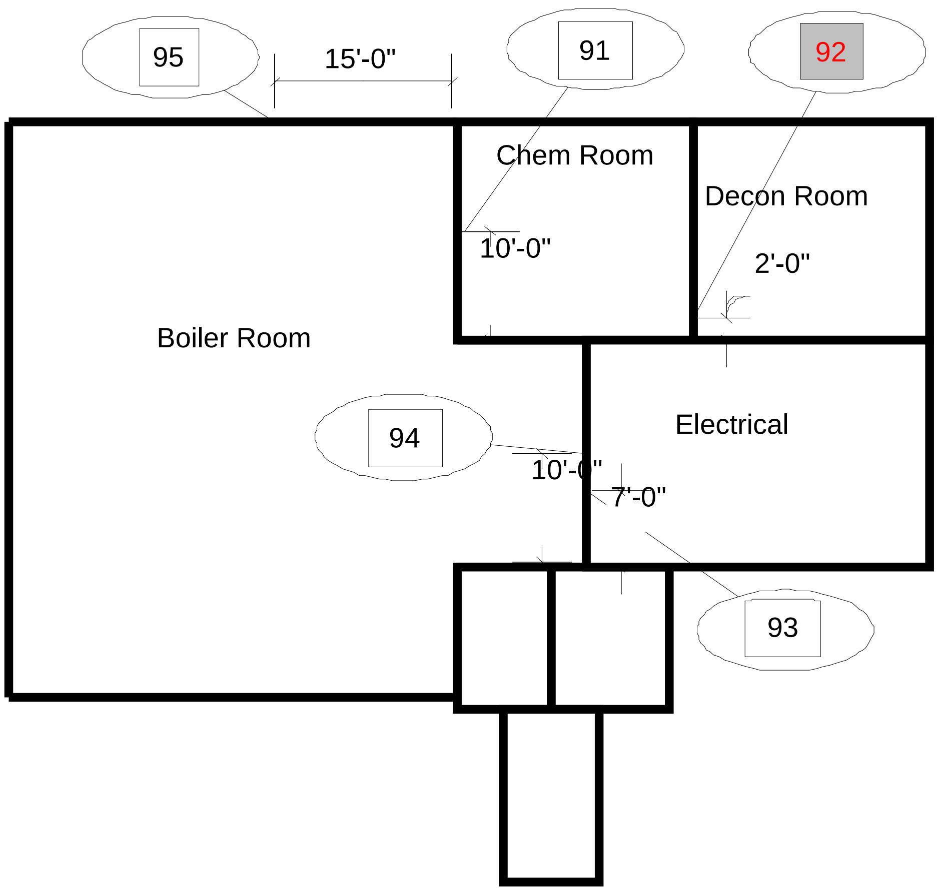


SAMPLING LOCATIONS

#	Energy*Solutions* , LLC Sampling Locations
#	Samples Selected by ORISE for Radiological Analyses

Sample Figure 3: Energy*Solutions* , LLC— Sampling Locations

CINDERBLOCK SAMPLING LOCATIONS



91- Inside Chem Room
92- Inside Decon Room
93- Inside Electrical Room
94- Inside Boiler Room
95- Outside Boiler Room

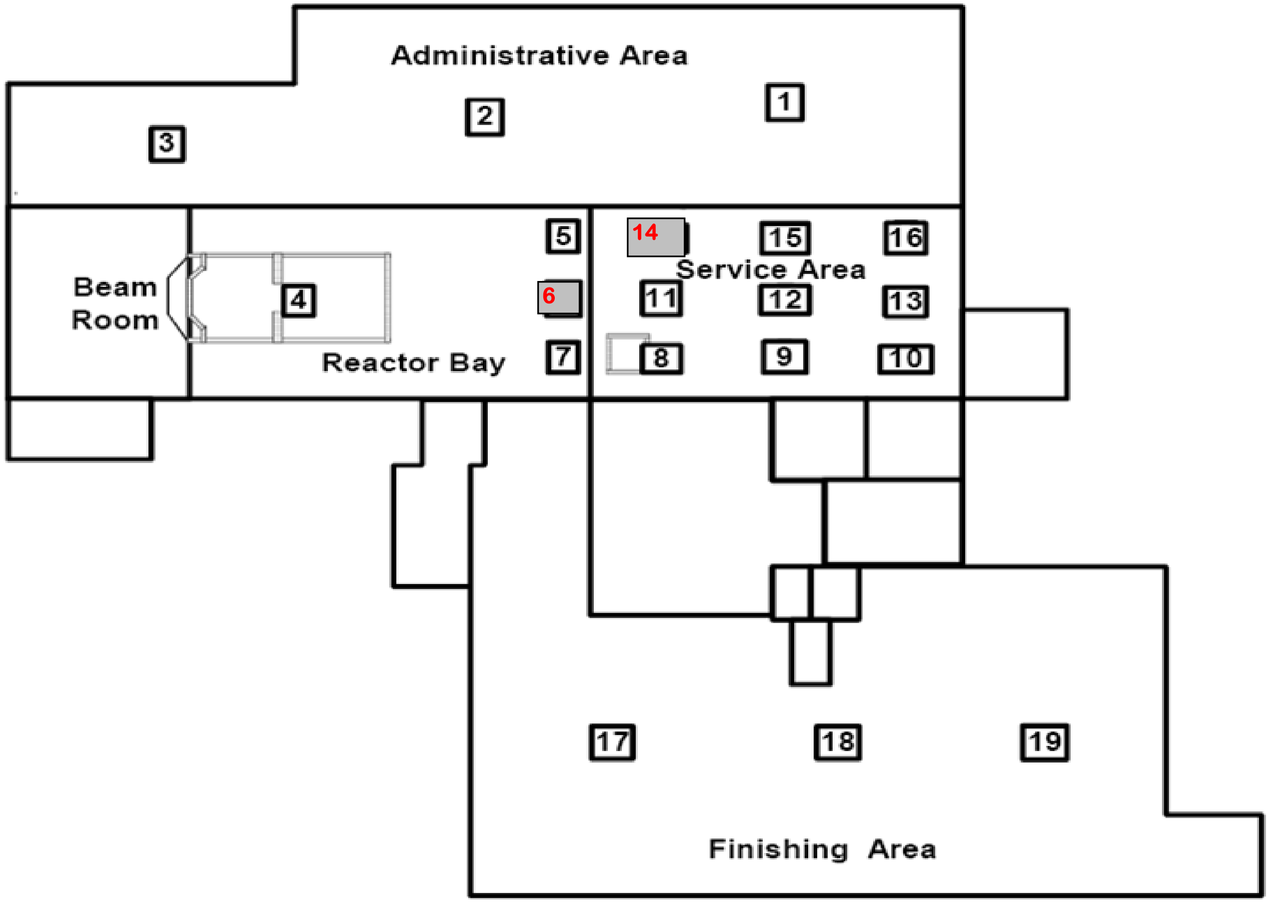
All heights at 3 feet.

SAMPLING LOCATIONS

— Energy*Solutions* , LLC Sampling Locations

— Samples Selected by ORISE for Radiological Analyses

Sample Figure 4: Energy*Solutions* , LLC— Sampling Locations



SAMPLING LOCATIONS

\# EnergySolutions, LLC Sampling Locations

\# Samples Selected by ORISE for Radiological Analyses

Sample Figure 5: Energy*Solutions*, LLC— Sampling Locations

APPENDIX B

MAJOR INSTRUMENTATION

APPENDIX B

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.

SCANNING INSTRUMENT/DETECTOR COMBINATIONS

Beta

Ludlum Floor Monitor Model 239-1
combined with
Ludlum Ratemeter-Scaler Model 2221
coupled to
Ludlum Gas Proportional Detector Model 43-37, Physical Area: 550 cm^2
(Ludlum Measurements, Inc., Sweetwater, TX)

Ludlum Ratemeter-Scaler Model 2221
coupled to
Ludlum Gas Proportional Detector Model 43-68, Physical Area: 126 cm^2
(Ludlum Measurements, Inc., Sweetwater, TX)

Ludlum Ratemeter-Scaler Model 2221
coupled to
Eberline Geiger-Muller (GM) Detector
Model HP-260, Physical Probe Area, 20 cm^2
(Eberline, Sante Fe, NM)

Gamma

Ludlum Pulse Ratemeter Model 12
(Ludlum Measurements, Inc., Sweetwater, TX)
coupled to
Victoreen NaI Scintillation Detector Model 489-55, Crystal: 3.2 cm x 3.8 cm
(Victoreen, Cleveland, OH)

LABORATORY ANALYTICAL INSTRUMENTATION

Low Background Gas Proportional Counter
Model LB-5100-W
(Tennelec/Canberra, Meriden, CT)

LABORATORY ANALYTICAL INSTRUMENTATION (CONTINUED)

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

High Purity Extended Range Intrinsic Detector
Model No. GMX-45200-5
(AMETEK/ORTEC, Oak Ridge, TN)
used in conjunction with:
Lead Shield Model SPG-16-K8
(Nuclear Data)
Multichannel Analyzer
DEC ALPHA Workstation
(Canberra, Meriden, CT)

High-Purity Germanium Detector
Model GMX-30-P4, 30% Eff.
(AMETEK/ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and
Multichannel Analyzer
DEC ALPHA Workstation
(Canberra, Meriden, CT)

APPENDIX C

SURVEY PROCEDURES

APPENDIX C

SURVEY PROCEDURES

PROJECT HEALTH AND SAFETY

Pre-survey activities included the evaluation and identification of potential health and safety issues. Tripping hazards over building debris and other materials in the facility were of particular concern for the indoor area surveys. Survey work was performed per the ORISE generic health and safety plans and a site-specific integrated safety management (ISM) pre-job hazard checklist which was completed and discussed with field personnel. Energy*Solutions*, LLC (ESL) also provided site-specific safety awareness training. All survey activities were conducted in accordance with ORISE health and safety and radiation protection procedures.

QUALITY ASSURANCE

Field survey activities were conducted in accordance with procedures from the following documents:

- Survey Procedures Manual (August 7, 2006)
- Laboratory Procedures Manual (April 18, 2006)
- Quality Program Manual (March 1, 2007)

The procedures contained in these manuals were developed to meet the requirements of the U.S. Department of Energy (DOE) Order 414.1C and the U.S. Nuclear Regulatory Commission *Quality Assurance Manual for the Office of Nuclear Material Safety and Safeguards* 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 MAPEP, NRIP, and ITP Laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures
- Periodic internal and external audits.

CALIBRATION PROCEDURES

Calibration of all field and laboratory instrumentation was based on standards/sources, traceable to the National Institute of Standards and Technology (NIST), when such standards/sources were available. In cases where they were not available, standards of an industry-recognized organization were used.

Detectors used for assessing surface activity were calibrated in accordance with ISO-7503[1] recommendations. The total efficiency (ε_{total}) was determined for each instrument/detector combination and consisted of the product of the 2π instrument efficiency (ε_i) and surface efficiency (ε_s): $\varepsilon_{total} = \varepsilon_i \times \varepsilon_s$.

ORISE selected Sr-90 as the beta calibration source (maximum beta energy of 1410 keV) as it provides a conservative representation of the primary beta emitters (Co-60 and Sr-90) and since the release criteria was based on Sr-90 as per the DP and FSSP. ISO-7503 recommends an ε_s of 0.25 for beta emitters with a maximum energy of less than 0.4 MeV (400 keV) and an ε_s of 0.5 for maximum beta energies greater than 0.4 MeV. Since the maximum beta energy for the chosen QDP facility calibration source was greater than 0.4 MeV, an ε_s of 0.5 was used to calculate ε_{total}.

Surface Scans

Hand-held detectors were placed on contact with the calibration sources. A postulated hot-spot size of 100 cm^2 was assumed *a priori* for determining scanning instrument efficiencies. The beta scanning Sr-90 ε_i value was 0.087 for the Geiger-Muller (GM) detectors and 0.44 for the hand-held gas proportional detectors; the calculated scanning Sr-90 ε_{total} value was 0.05 for the GM detectors and 0.22 for the hand-held gas proportional detectors[2]. For the calibration source, emission rates were not corrected for geometry when sources larger than the detectors were used.

The scanning ε_{total} was determined for the floor monitor in the same fashion as above for the hand-held gas proportional detectors with the exception that typical efficiencies for the floor monitor were used for these survey activities rather than specific calibration efficiencies. For the floor monitor, the scanning ε_i for Sr-90 was 0.42; the scanning ε_{total} was 0.21[2].

---

[1]International Standard. ISO 7503-1, Evaluation of Surface Contamination - Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters. August 1, 1988.
[2] Decommissioning Health Physics: A Handbook for MARSSIM Users. E.W. Abelquist. Institute of Physics. 2001.

Surface Activity Measurements

The calibration ε_i values for the GM and hand-held gas proportional detectors used for the confirmatory survey were 0.64 and 0.66 for Sr-90, respectively. Calibration source emission rates were corrected to the active area of the detector when the calibration source area exceeded the detector area. The static Sr-90 ε_{total} values used were 0.32 for the GM detector and 0.33 for the gas proportional detector.

SURVEY PROCEDURES

Surface Scans

Structural 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 with a 0.8 milligram per square centimeter (mg/cm^2) window and a NaI scintillation detector were used to scan the floors of the surveyed areas. Wall surfaces were scanned using small area hand-held gas proportional (126 cm^2) detectors with a 0.8 mg/cm^2 window and GM (20 cm^2) detectors. Identification of elevated levels was based on increases in the audible signal from the recording and/or indicating instrument.

Scan minimum detectable concentrations (MDCs) were estimated using the calculational approach described in NUREG-1507[3]. The scan MDC is a function of many variables, including the background level. Site surface activity background levels were within the typical range of 800 to 1,400 counts per minute (cpm) for the large area gas proportional detectors (floor monitors) and 200 to 450 cpm for the hand-held gas proportional detectors. The hand-held gas proportional background for surface activity was re-determined on site and was 248 cpm; the GM background was 60 cpm. Additional parameters selected for the calculation of scan MDC included a one-second observation interval, a specified level of performance at the first scanning stage of 95% true positive rate and 25% false positive rate, which yields a d′ value of 2.32 (NUREG-1507, Table 6.1), and a surveyor efficiency of 0.5. To illustrate an example for the hand-held gas proportional detectors with 0.8 mg/cm^2 windows, the minimum detectable count rate (MDCR) and scan MDC can be calculated as follows:

[3] NUREG-1507. Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions. US Nuclear Regulatory Commission. Washington, DC; June 1998.

b_i = (248 cpm) (1 s) (1 min/60 s) = 4.13 counts

MDCR = (2.32) (4.13 counts)$^{1/2}$ [(60 s/min) / (1 s)] = 283 cpm

$MDCR_{surveyor}$ = 283 / (0.5)$^{1/2}$ = 400 cpm

The scan MDC is calculated using the total scanning efficiency (ε_{total}) of 0.22:

$$\text{Scan MDC} = \frac{MDCR_{surveyor}}{\varepsilon_{total}} \text{ dpm/100 cm}^2$$

The scan MDC for the hand-held gas proportional detector was calculated to be 1,820 dpm/100 cm^2; the scan MDC for the GM detector using the same calculational approach was 3,940 cpm. For the given floor monitor background ranges, the scan MDC ranged from 3,420 to 4,530 dpm/100 cm^2.

Specific scan MDCs for the NaI scintillation detector for Co-60 and Cs-137 in concrete were not determined as the instrument was used solely as a qualitative means to identify elevated gamma activity. MDCs for radionuclides in the concrete would approximate those contained in NUREG-1507 which are 5.8 and 10.4 pCi/g, respectively.

Surface Activity Measurements

Measurements of total beta surface activity levels were performed using hand-held gas proportional and GM detectors coupled to portable ratemeter-scalers. Count rates (cpm), which were integrated over one minute with the detector held in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the count rate by the total static efficiency ($\varepsilon_i \times \varepsilon_s$) and correcting for the physical area of the detector. ORISE did not determine construction material-specific background for each surface type encountered for determining net count rates. Instead, ORISE took a conservative approach and did not subtract material specific backgrounds in determining surface activity levels. At the request of the NRC, ORISE also determined the uncertainties for the direct measurement results. The single-point 95% confidence level uncertainties were calculated as follows:

$$2\sigma = 2 \times \frac{\sqrt{Counts + BKG}}{T\,\varepsilon_T\,G}$$

where, σ = standard deviation of the count

T = time (min) (same count time for Counts and BKG)

ε_T = total efficiency

G = geometry factor

BKG = background counts

Counts = gross activity counts (source plus background)

Surface activity measurements were performed on concrete, brick, metal, and wood. The static surface activity MDC was 185 dpm/100 cm^2 for the gas proportional detector and 609 dpm/100 cm^2 for the GM detector. The physical surface areas assessed by the gas proportional and GM detectors were 126 and 20 cm^2, respectively.

Miscellaneous Sampling

Concrete bore and metal roof samples were collected by Energy*Solutions* personnel. These samples were placed in plastic bags and sealed. ORISE selected several samples and labeled them in accordance with ORISE survey procedures.

RADIOLOGICAL ANALYSIS

Gross Beta

Smears were counted for two minutes on a low-background gas proportional system for gross beta activity. The MDC of the gross beta procedure was 15 dpm/100 cm^2.

Gamma Spectrometry

Miscellaneous (concrete bore and metal roof) samples were placed in an appropriate container. The container was placed approximately 10 cm above the detector in an air filter geometry to minimize the affect of the sample quantity. Samples of concrete were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in an appropriate container—the quantity placed in the container was chosen to reproduce the calibrated counting geometry. Net material weights were determined and the samples counted using intrinsic germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system.

All total absorption peaks (TAP) associated with the radionuclides-of-concern were reviewed for consistency of activity. TAPs used for determining the activities of radionuclides of concern and the typical associated MDCs for a one-hour count time were:

Radionuclide	TAP (MeV)	MDC (pCi/g)
Co-60	1.173	0.05
Cs-137	0.662	0.05

Spectra were also reviewed for other identifiable TAPs.

Strontium Analyses

Solid samples were ashed and dissolved as necessary. Samples with high calcium concentrations had carriers and ethylenediaminetetraacetate (Na_2EDTA) added and were passed through a cation exchange resin. Alkali metals and most alkaline earths were absorbed on the cation resin, and the complexed calcium passed through unabsorbed. Alkaline earth metals were removed from the cation resin by elution with a sodium chloride solution and precipitated as carbonates. Barium was removed by chromate precipitation.

Strontium concentrations within the samples were then determined in a low-background gas proportional counter, and the count rate was corrected for yttrium ingrowth. The chemical yield was determined gravimetrically. The typical MDC of the procedure is 2 pCi/g wet weight for concrete.

DETECTION LIMITS

The uncertainties associated with the analytical data presented in the tables of this report represent the total propagated uncertainties for that data. These uncertainties were calculated based on both the gross sample count levels and the associated background count levels.

Detection limits, referred to as minimum detectable concentration (MDC), were based on 3 plus 4.65 times the standard deviation of the background count $[3 + (4.65\ (BKG)^{1/2})]$. 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. The uncertainties associated with the direct measurement data presented in the tables of this report were calculated based on counting statistics only.