UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

September 24, 2008

Mr. James Shetler, Assistant General Manager
Energy Supply
Sacramento Municipal Utility District
6201 'S' Street
P.O. Box 15830
Sacramento, California 95852

SUBJECT: NRC INSPECTION REPORT 050-00312/08-003

Dear Mr. Shetler:

An inspection was conducted on September 8 -11, 2008, by the Nuclear Regulatory Commission (NRC) at your Rancho Seco Nuclear Generating Station. At the end of the inspection on September 11, 2008, the inspector briefed the Manager, Plant Closure and Decommissioning (Plant Manager), regarding the preliminary inspection findings. The enclosed inspection report presents the scope and results of the inspection.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection included an assessment of your safety reviews, design changes, and modifications; decommissioning performance; final status surveys; and radioactive waste treatment, effluent, and environmental monitoring. The inspection determined that you were conducting decommissioning activities in compliance with license and regulatory requirements. No violations were identified, and no response to this letter is required.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/Adams.html. To the extent possible, your response, if you chose to make one, should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction.

Should you have any questions concerning this inspection, please contact Mr. Emilio Garcia, Health Physicist, at (530) 756-3910 or the undersigned at (817) 860-8197.

Sincerely,

Jack E. Whitten, Chief
Nuclear Materials Safety Branch B

Docket No.: 050-00312
License No.: DPR-54

Enclosure:
NRC Inspection Report 050-00312/08-003
(w/Attachment)

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	050-00312
License No.:	DPR-54
Report No.:	050-00312/08-003
Licensee:	Sacramento Municipal Utility District
Facility:	Rancho Seco Nuclear Generating Station
Location:	14440 Twin Cities Road Herald, California
Dates:	September 8-11, 2008
Inspector:	Emilio M. Garcia, Health Physicist
Approved By:	Jack E. Whitten, Chief Nuclear Materials Safety Branch B
Attachments:	Supplemental Information

Enclosure

EXECUTIVE SUMMARY

Rancho Seco Nuclear Generating Station
NRC Inspection Report 050-00312/08-003

This inspection was a routine, announced inspection of decommissioning activities being conducted at the Rancho Seco Nuclear Generating Station. Areas inspected included safety reviews, design changes, and modifications, decommissioning performance, final status surveys, and radioactive waste treatment, effluent, and environmental monitoring.

Safety Reviews, Design Changes, and Modifications

- Safety evaluations were conducted in accordance with the licensee's procedures and applicable regulations. Training of safety screen reviewers, and Commitment Management Review Group members and alternates, met applicable requirements (Section 1).

Decommissioning Performance and Status Review

- The licensee continued to remediate contaminated surfaces in a safe manner. Final status surveys had been completed on 278 of a projected 315 survey units (Section 2).

Inspection of Final Status Surveys

- The Oak Ridge Institute for Science and Education (ORISE) staff conducted confirmatory measurements on selected surfaces of the auxiliary building spent fuel building and the turbine building. The results of the ORISE surveys conducted during this inspection will be reported at a later date (Section 3).

- The results of survey activities conducted by ORISE staff during the April 14 through 17 and May 29, 2008, site inspections were documented by ORISE in a report issued on August 29, 2008. A copy of this ORISE report is attached as Enclosure 2. This report concludes that for those areas surveyed by ORISE results were in agreement with the licensee's reported values and did not exceed for any survey unit the specific designed derived concentration guidelines elevated measurement comparison criteria (Section 3).

Radioactive Waste Treatment, Effluent and Environmental Monitoring

- The licensee had revised the offsite dose calculation manual to reflect the condition of the plant and to address operational needs, but these modifications had not diminished the licensee's ability to calculate radiological doses offsite (Section 4.1).

- The licensee did not have any liquid effluent pathways and therefore no liquid process or effluent monitors. Particulate airborne effluents were continuously sampled from the Reactor Building and from the Interim Onsite Storage Building stack (Section 4.2).

- The Annual Radiological Environmental Operating Reports for 2007 was submitted on a timely basis and met applicable requirements. Radioactivity levels in the sampled media were consistent with previous years and were below the NRC-required reportable levels (Section 4.3).

- The 2007 Annual Radioactive Effluent Release Report was submitted timely and met applicable requirements. Releases of radioactivity in gaseous and liquid effluents in 2007 did not exceed applicable limits (Section 4.4).

Report Details

Summary of Facility Status

The Rancho Seco Nuclear Generating Station (Ranch Seco) was permanently shut down in June 1989. All spent reactor fuel has been moved to an onsite Independent Spent Fuel Storage Installation. At the time of this inspection, the licensee was conducting decommissioning activities under the provisions of the incremental decommissioning option of Rancho Seco's Post Shutdown Decommissioning Activities Report dated March 20, 1997.

Decommissioning conducted by the licensee included work activities in the auxiliary building, reactor building, spent fuel building, and exterior areas. All major components had been removed, packaged, and shipped offsite for disposal. In the auxiliary building, remediation was completed and only one room remained for final status survey. In the reactor building, the concrete and steel removal project had been completed, including the removal of the reactor building polar crane. Remediation of surfaces continued in the reactor building. In the spent fuel building, most of the remediation of the surfaces had been completed. At the time of the inspection, the licensee had completed final status surveys on approximately 88 % of all survey units.

1 Safety Reviews, Design Changes, and Modifications (IP 37801)

1.1 Inspection Scope

The inspector reviewed selected 10 CFR 50.59 safety evaluations conducted since the previous inspection of this program area.

1.2 Observations and Findings

The inspector observed that the licensee had not conducted a design change to the facility since this area was last inspected in December 2007. Four full 10 CFR 50.59 evaluations had been performed since the last inspection. These evaluations included: Revisions 21 and 22 to the Offsite Dose Calculation Manual (ODCM) Procedure CAP-002, Revision 32 to the Nuclear Organization Procedure RSAP-0101, and Revision 1 to the License Termination Plan. The licensee determined, and the inspector confirmed, that none of these procedure revisions would require prior NRC approval before implementation. The inspector reviewed records of the licensee's Commitment Management Review Group (CMRG) generated since the last inspection. The CMRG meeting minutes reviewed by the inspector indicated that the safety evaluation packages had been reviewed, discussed, and unanimously approved. Training records maintained by the licensee indicated that all of the CMRG members and alternates had been trained as qualified 10 CFR 50.59 reviewers.

The inspector also reviewed the safety screening packages for the 17 procedure revisions that the licensee concluded did not require a full safety evaluation. The inspector determined that these packages were complete and had been reviewed in accordance with 10 CFR 50.59 requirements. Both a qualified reviewer and a second level reviewer had signed the packages. The inspector confirmed that all safety screening package reviewers were on the list of qualified reviewers maintained by the licensee. Training records reviewed by the inspector indicated that these individuals had successfully completed initial training as a 10 CFR 50.59 reviewer or they had received refresher training within the last 12 months.

1.3 Conclusion

Safety evaluations were conducted in accordance with the licensee's procedures and applicable regulations. Training of safety screen reviewers, and CMRG members and alternates, met applicable requirements.

2 Decommissioning Performance and Status Review (IP 71801)

2.1 Inspection Scope

The inspectors interviewed personnel, reviewed selected documents, and toured portions of the site to observe decommissioning work activities including housekeeping, safety practices, fire protection practices, and radiological controls.

2.2 Observations and Findings

The inspector conducted tours of the reactor, auxiliary, spent fuel, turbine and Interim Onsite Storage (IOSB) buildings and observed decommissioning activities in progress. Decommissioning work observed by the inspector during the tour was being conducted in a safe and orderly manner. The inspector conducted independent radiological surveys using a Ludlum Model 2401-P survey meter (NRC No. 21190G, calibration due date September 21, 2008). Radiological controls, including postings and barriers, were observed by the inspector to be in place.

The reactor building concrete and steel removal project that included the removal of the polar crane had been completed. At the time of this inspection, the licensee was conducting remediation of contaminated liner surfaces. The licensee indicated that a 12-foot section of liner plate and the activated concrete below the liner plate where the reactor vessel had resided remained to be removed. As of the September 10, 2008, 7 of 36 polar crane rail supports remained to be remediated. The licensee informed the inspector that they also plan to removing from inside the inside the reactor building approximately 3 feet of the emergency sump lines that had contamination above the DCGLs.

In the auxiliary building, all rooms except one had been remediated and the final status surveys for each room had been completed. At the time of the inspection, the ventilation equipment room and room 208/211 was undergoing final status surveys. The licensee, when question by the inspector about the completion schedule, projected the final status survey of the ventilation equipment room would be completed by September 30, 2008.

All remediation and final status surveys had been performed in the turbine building, except for the high pressure turbine pedestal number 1, which the licensee projected would be completed by September 12, 2008.

In the spent fuel building, remediation and final status surveys had been completed in all but a small section below the cask loading platform. The licensee projected that it would complete remediation and final status surveys of the spent fuel building by September 30, 2008.

As of September 10, 2008, the licensee had completed final status surveys on 278 of a projected 315 survey units. The number of projected survey units had changed since the last inspection as some units were split by the licensee into multiple units. This completion of 278 survey units constituted approximately 88 % of the projected survey units. The licensee projected that it would complete remediation and final status surveys of the site by October 31, 2008.

2.3 Conclusion

The licensee continued to remediate contaminated surfaces in a safe manner. Final status surveys had been completed on 278 of a projected 315 survey units constituting approximately 88 % of the projected survey units.

3 Inspection of Final Surveys (IP 83801)

3.1 Inspection Scope

Independent confirmatory radiological measurements were performed by Oak Ridge Institute for Science and Education (ORISE) on surfaces of the spent fuel building.

3.2 Observations and Findings

On April 12, 2006, the licensee submitted their License Termination Plan (LTP) to the NRC. This LTP included proposed designed derived concentration guidelines (DCGLs) for meeting the public dose limits after license termination. On November 27, 2007, the NRC issued License Amendment Number 133 that approved the licensee's LTP and the respective DCGLs.

Representatives from ORISE, working as the NRC's contractor, reviewed records of final status surveys taken in the auxiliary, spent fuel, and the turbine buildings. At the NRC's request, ORISE personnel conducted independent confirmatory radiological measurements of selected locations in the auxiliary, spent fuel, and the turbine buildings and compared their survey results with the licensee's survey results. The results of these surveys taken by ORISE will be reported to the licensee at a later date under separate correspondence.

The results of survey activities conducted by ORISE staff during the April 14 through 17 and May 29, 2008, site inspections were documented in a report issued on August 29, 2008. A copy of that report is attached. This report concludes that for the areas surveyed the results obtained by ORISE were in agreement with the licensee's reported values. The report notes that of the 70 locations where direct measurements were performed, 15 exceeded the site-specific gross beta DCGLs. However, all survey units were within the specific designed DCGL elevated measurement comparison criteria.

On September 11, 2008, when preparing a decommissioning technical basis document the license identified an error that they had made in calculating the surrogate DCGLs. The guidance document Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) specified that a ratio of the concentration of the hard-to-detect (HTD) nuclide to the designated surrogate nuclide be used when establishing the surrogate DCGL. The licensee concluded that they had used a ratio of the HTD nuclide to the sum of the concentrations of all surrogate nuclides present. This was determined by the licensee to be a non-conservative error. To resolve this non-conservative error, the licensee opened Potential Deviation from Quality (PDQ) 08-017 to review this problem and document the corrective actions that would be taken to resolve this issue.

The inspector noted that this problem had been identified by the licensee, will be addressed by the licensee prior to the termination of the license for this site, and concluded that no safety consequences had resulted from this propagated error.

3.3 Conclusion

The ORISE staff conducted confirmatory measurements on selected surfaces of the auxiliary, spent fuel, and the turbine buildings. The results of the ORISE surveys conducted during this inspection will be reported at a later date.

The results of survey activities conducted by ORISE staff during the April 14 through 17 and May 29, 2008, site inspections were documented in a report issued on August 29, 2008. A copy of this ORISE report is attached as Enclosure 2. This report concludes that for the areas surveyed by ORISE the results obtained were in agreement with the licensee's reported values and did not exceed the survey unit specific designed DCGLs elevated measurement comparison criteria.

4. Radioactive Waste Treatment, Effluent and Environmental Monitoring (IP 84750)

4.1 Changes in the Offsite Dose Calculation Manual

a. Inspection Scope

The inspector discussed the changes to the ODCM with the Radiation Health Supervisor and reviewed the current ODCM.

b. Observations and Findings

Chemistry Administrative Procedure CAP-0002, "Offsite Dose Calculation Manual," contained the methodology and parameters used in the calculating of off-site doses due to radioactive gaseous and liquid effluents. Revision 21 of this procedure was made effective on July 7, 2008, after the licensee completed the reactor building concrete and steel removal project. This revision made by the licensee reclassified the reactor building exhaust as a miscellaneous release and revised the applicable sections of the procedure.

On July 8, 2008, revision 22 to the ODCM was issued to remove the liquid effluent pathway from the Scope section of the procedure, and to make a number of other specified changes related to the reclassification of the reactor building exhaust as a miscellaneous release pathway.

The inspector concluded that these changes to the ODCM reflect the condition of the plant, addressed operational needs, and did not diminish the ability of the licensee to calculate offsite dose.

c. Conclusions

The licensee had revised the offsite dose calculation manual to reflect the condition of the plant and address operational needs. The inspector determined that the changes made by the licensee to the ODCM had not diminished their ability to calculate radiological doses offsite.

4.2 Process and Effluent Radiation Monitors

a. Inspection Scope

The inspector discussed the status of process and effluent monitors with the licensee's staff and reviewed related records.

b. Observations and Findings

At the time of the inspection Rancho Seco did not have any liquid effluent pathways and therefore no liquid process or effluent monitors.

With the completion of the reactor building concrete and steel removal project, on July 7, 2008, the reactor building exhaust was reclassified as a miscellaneous release pathway. A miscellaneous release pathway is defined by the licensee as a release pathway that is considered planned but that has no explicit monitoring requirements. Miscellaneous release pathways contribute to less than 5 % of annual dose limits. At the time of the inspection, and at the advanced state of decommissioning Rancho Seco is currently at, the Reactor Building and the IOSB are the only credible gaseous effluent pathways. Both of these facilities were considered miscellaneous release pathways.

Attachment 11 to Revision 22 of the ODCM specified that for both the Reactor Building and the IOSB continuous particulate sampling should take place with monthly analysis of the samples for principal gamma emitters, cobalt-60, cesium-134, and cesium-137. The inspector noted that the licensee was conducting weekly analysis of the continuous samples collected from the Reactor Building and the IOSB. Effluent release permits had been generated by the licensee for the Reactor Building; however, since no detectable activity had been found in the samples from the IOSB, no release permits had been generated for this release path.

On September 11, 2008, during the site tour, the inspector observed air samplers in the Reactor Building and the IOSB. The inspector observed three air samplers in the Reactor Building and one on the stack from the IOSB. These four air samplers were operational and stickers indicated they were in calibration.

c. Conclusions

The licensee did not have any liquid effluent pathways and therefore no liquid process or effluent monitors. Particulate airborne effluents were continuously sampled from the Reactor Building and from the IOSB stack.

4.3 Annual Radiological Environmental Operating Report for 2007

a. Inspection Scope

The 2007 Annual Radiological Environmental Operating Reports was reviewed.

b. Observations and Findings

Step 1.5.2.3 of Appendix A to the Rancho Seco Quality Manual (RSQM) requires that an Annual Radiological Environmental Operating Report covering the previous year be submitted to the NRC prior to May 1 of each year. On April 21, 2008, the licensee submitted the 2007 report. This report demonstrated that atmospheric, terrestrial and aquatic environments, and the land use adjacent to Rancho Seco Nuclear Station were being monitored as required. Radioactivity levels in the sampled media were observed by the inspector to be below the NRC required reportable levels. The Annual Radiological Environmental Operating Report concluded that Rancho Seco Nuclear Station had no significant radiological impacts on the environment.

c. Conclusions

The 2007 Annual Radiological Environmental Operating Report was submitted on a timely basis and met applicable requirements. Radioactivity levels in the sampled media were consistent with previous years and were below the NRC-required reportable levels.

4.4 Annual Radioactive Effluent Release Report for 2007

a. Inspection Scope

The 2007 Annual Radioactive Effluent Release Report was reviewed.

b. Observations and Findings

Step 1.5.3 of Appendix A to the RSQM required that an Annual Radioactive Effluent Release Report covering the previous 12 months be submitted to the NRC within 90 days of January 1 of each year. On March 31, 2008, the licensee submitted the 2007 Annual Radioactive Effluent Release Report. The report provided to the NRC included summaries of radioactive gaseous and liquid releases from the site. The report also concluded that the releases of radioactivity in gaseous and liquid effluents had not exceeded the limits of 10 CFR 20 or the numerical guidelines of 10 CFR 50, Appendix I. The licensee concluded that a 40 CFR 190 dose evaluation was not required because radioactive effluent releases specified in the Annual Radioactive Effluent Release Report did not exceed twice the numerical guidelines of 10 CFR 50, Appendix I.

The licensee reported in the Annual Radioactive Effluent Release Report that there were no unplanned gaseous or liquid releases during 2007. There were 3 planned and monitored liquid releases in 2007.

Technical Requirement 6.12.3 of the licensee's ODCM identified effective dose commitment limits from liquid effluents to members of the public at or beyond the site boundary. These effective dose commitment limits were based on the numerical guidelines specified in 10 CFR 50, Appendix I. Appendix I establishes limits of 3 millirem per calendar year to the total body or 10 millirem to any organ. The 2007 Annual Radioactive Effluent Release Report calculated total effective dose due to liquid effluents to be 1.04E-01 millirem or approximately 4 % of the applicable limit specified in 10 CFR Part 50, Appendix I. The licensee's report specified a maximum calculated annual organ dose commitment of 3.96 E-02 millirem or approximately 4 % of the applicable limit specified in 10 CFR Part 50, Appendix I.

Technical Requirement 6.12.7 of the licensee's ODCM specifies effective dose commitment limits from gaseous effluents to members of the public at or beyond the site boundary. These limits adopted by the licensee were based on the numerical guidelines of 10 CFR 50, Appendix I, that specifies for Tritium and radioactive material in particulate form with half-lives greater than 8 days is 7.5 millirem per calendar quarter for the total body or 15-millirem per calendar year to any organ. During 2007 there were no airborne releases of noble gases. The annual calculated organ dose at the site boundary due to tritium and radioactive material in particulate form was 1.94E-02 millirem or approximately 0.2 % of the applicable limit specified in 10 CFR Part 50, Appendix I.

During 2007, no direct radiation attributable to the plant was recorded by radiation dosimetry monitoring badges.

In 2007, there were 286 shipments of solid radioactive waste made. All solid waste shipments were transported by highway or rail and all shipments went to a licensed low-level radioactive waste disposal facility. Based on the information provided, the inspector calculated that the total volume of waste shipped by the licensee was approximately 8170 cubic meters with a total activity of approximately 79.8 curies.

c. <u>Conclusions</u>

The 2007 Annual Radioactive Effluent Release Report was submitted in a timely manner and met applicable requirements. Releases of radioactivity in gaseous and liquid effluents in 2007 did not exceed applicable limits.

5 Exit Meeting Summary

At the end of the site visit on September 11, 2008, the inspector briefed the Plant Closure and Decommissioning (Plant Manager) and other members of licensee staff regarding the preliminary inspection findings. The licensee did not identify any information provided to, or reviewed by, the inspector as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

<u>Sacramento Municipal Utility District</u>

M. Bua, Radiation Protection/Chemistry Superintendent
R. Decker, Lead Final Status Surveys Engineer
J. Field, Engineering Superintendent
W. Hawley, Dismantlement Superintendent - Operations
L. Hoist, Nuclear Document Control Supervisor
R. Jones, Supervising Quality Engineer
M. Murdock, Field Oversight Decontamination Engineer
S. Nicolls, Radiation Health Supervisor
S. Redeker, Manager, Plant Closure and Decommissioning (Plant Manager)
G. Roberts, Maintenance Superintendent
E. Ronningen, Superintendent Rancho Seco Assets

INSPECTION PROCEDURES USED

IP 37801	Safety Reviews, Design Changes, and Modifications
IP 71801	Decommissioning Performance and Status Review
IP 83801	Inspections of Final Surveys
IP 84750	Radioactive Waste Treatment, Effluents and Environmental Monitoring

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

None

<u>Closed</u>

None

<u>Discussed</u>

None

LIST OF ACRONYMS

CFR	Code of Federal Regulations
CMRG	Commitment Management Review Group
DCGLs	Derived Concentration Guide Lines
HTD	Hard-To-Detect
IOSB	Interim Onsite Storage Building
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
ODCM	Offsite Dose Calculation Manual
ORISE	Oak Ridge Institute for Science and Education
PDQ	Potential Deviation from Quality
RSAP	Rancho Seco Administrative Procedure
RSQM	Rancho Seco Quality Manual

ORISE

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

August 29, 2008

Mr. John Hickman
Mail Stop: T-8F5
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and Environmental Management Programs
U.S. Nuclear Regulatory Commission
11545 Rockville Pike
Rockville, MD 20852

SUBJECT: DOE CONTRACT NO. DE-AC05-06OR23100
REVISED INTERIM LETTER REPORT - CONFIRMATORY SURVEY RESULTS FOR ACTIVITIES PERFORMED IN APRIL AND MAY 2008
RANCHO SECO NUCLEAR GENERATING STATION
HERALD, CALIFORNIA
DCN 1695-SR-03-1 (DOCKET NO. 50-312, RFTA NO. 06-003)

Dear Mr. Hickman:

The Oak Ridge Institute for Science and Education (ORISE) performed confirmatory survey activities on the Auxiliary Building structural surfaces (Rooms 10, 15, 40, 42, 51 and 52), the Pump Alley, the Fuel Storage Building exterior excavation, and portions of the Rad Waste and Acid Waste drain systems at the Rancho Seco Nuclear Generating Station in Herald, California on April 14 through 17 and May 29, 2008. These survey activities were requested and approved by the U.S. Nuclear Regulatory Commission (NRC). Enclosed is a revised interim letter report to replace the report, issued on July 24, 2008, that contained figures that were misprinted. This revised report summarizes ORISE's survey procedures and preliminary results of the confirmatory survey. The surveys included beta and gamma surface scans, direct measurements for total net beta activity, and smears for removable alpha and beta activity within the Auxiliary Building; embedded piping gamma scans within the Auxiliary Building; and limited gamma scans and the collection of soil samples in the Pump Alley and the Fuel Storage Building exterior excavation.

If you have any questions, please direct them to me at 865.576.0065 or Tim Vitkus at 865.576.5073.

Sincerely,

Wade C. Adams
ORISE Health Physicist/Project Leader
Survey Projects

WCA:km

Enclosure

c: T. Carter, NRC/FSME/DWMEP/DD/SP T-8F5
E. Knox-Davin, NRC/FSME/TWFN 8A23
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REVISED INTERIM LETTER REPORT
CONFIRMATORY SURVEY RESULTS
FOR ACTIVITIES PERFORMED IN APRIL AND MAY 2008
RANCHO SECO NUCLEAR GENERATING STATION
HERALD, CALIFORNIA

Prepared by

W. C. Adams

Prepared for the
U.S. Nuclear Regulatory Commission

AUGUST 2008

ORISE

This report is based on worked performed by the Oak Ridge Institute for Science and Education under contract number DE-AC05-06OR23100 with the U.S. Department of Energy.

REVISED INTERIM LETTER REPORT
CONFIRMATORY SURVEY RESULTS
FOR ACTIVITIES PERFORMED IN APRIL AND MAY 2008
RANCHO SECO NUCLEAR GENERATING STATION
HERALD, CALIFORNIA

INTRODUCTION

The Sacramento Municipal Utility District (SMUD) operated the Rancho Seco Nuclear Generating Station (RSNGS) from 1976 to 1989 under Atomic Energy Commission Docket Number 50-312 and License Number DPR-54. In August 1989, SMUD notified the U.S. Nuclear Regulatory Commission (NRC) of the RSNGS's permanent shutdown. In May 1991, SMUD submitted the Rancho Seco Decommissioning Plan (DP) which was approved by the NRC in March 1995. SMUD began decommissioning activities in February 1997 and completed transfer of all the spent nuclear fuel in August 2002 (SMUD 2006a).

In April 2006, SMUD submitted a license termination plan (LTP) that the NRC approved on November 26, 2007 (SMUD 2006a and NRC 2007). SMUD is currently conducting decontamination efforts and performing final status surveys (FSS) on the remaining structural surfaces and in open land areas.

The NRC requested that the Independent Environmental Assessment and Verification (IEAV) Program of the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory surveys of structural surfaces in several Auxiliary Building rooms, the Pump Alley, the Acid Waste System and Rad Waste System drains in the Auxiliary Building and the Fuel Storage Building exterior excavation at the RSNGS (Figures 1 and 2). The confirmatory surveys were performed during the period of April 14 through 17 and May 29, 2008.

PROCEDURES

Confirmatory surveys were performed in accordance with a site-specific survey plan that was submitted to and approved by the NRC (ORISE 2007a). The site-specific survey plan follows the guidance provided in the IEAV Survey Procedures and Quality Program Manuals (ORISE 2008a and ORAU 2007).

In the Auxiliary Building, ORISE performed confirmatory survey activities on structural surfaces in Rooms 10, 15, 40, 42, 51, 52, the 51 and 52 Excavation, and in the Pump Alley (Figures 3 through 15). ORISE performed survey activities on the interior surfaces within two Acid Waste System drains (Figures 16 and 17) and two Rad Waste System drains (Figure 18). At the request of the NRC site representative, ORISE also performed confirmatory radiological surveys of the Fuel Storage Building exterior excavation (Figure 19).

SURFACE SCANS

<u>Auxiliary Building Structural Surfaces</u>

Gamma surface scans were performed using sodium iodide, thallium-activated [NaI(Tl)] gamma scintillation detectors coupled to ratemeters with audible indicators. Beta surface scans were performed using large area gas proportional, hand-held gas proportional, and Geiger-Muller (GM) detectors coupled to ratemeter-scalers with audible indicators. Particular attention was given to cracks, joints, embedded piping openings and horizontal surfaces in the evaluated structural surfaces where material may have accumulated.

Drains and Pipe Penetrations

Limited qualitative gamma scans were performed in portions of two Acid Waste System drain lines and in two Rad Waste System drain lines. Gamma scans were performed using a cesium iodide, thallium-activated [CsI(Tl)] gamma scintillation detector coupled to a ratemeter with an audible indicator. The detector was also paused at each one-foot scan increment and the gamma count rate recorded. These data were used for direct comparison with the SMUD FSS data.

Pump Alley and Fuel Storage Building Exterior Excavation Areas

Gamma scans of the soils within the excavated Pump Alley and the Fuel Storage Building exterior excavation were performed using a NaI(Tl) gamma scintillation detector coupled to a ratemeter with an audible indicator. Beta surface scans of the Pump Alley concrete side walls were performed using hand-held gas proportional detectors coupled to ratemeter-scalers with audible indicators.

SURFACE ACTIVITY MEASUREMENTS

Based on beta and gamma surface scan results, direct measurements for beta activity were performed at 59 judgmentally-selected locations on the evaluated structural surfaces within the Auxiliary Building which were available for confirmatory survey activities; 11 direct measurement locations were performed within the Pump Alley. Direct measurements were performed using hand-held gas proportional detectors coupled to ratemeters-scalers. A smear sample for determining removable gross alpha and gross beta activity levels was collected from each direct measurement location. Direct measurement and smear locations are indicated on Figures 3 through 14.

SOIL SAMPLING

ORISE collected four soil samples based on gamma scan results during the initial confirmatory survey activities of the Pump Alley soil excavation in April 2008. SMUD personnel performed gamma spectroscopy on several of the samples and determined that the soil derived concentration guideline levels (DCGLs) were exceeded and that further remedial actions would be required. ORISE returned the four Pump Alley soil samples to SMUD personnel as additional confirmatory survey activities of the Pump Alley would be required. ORISE collected two judgmental soil samples from the Fuel Storage Building exterior excavation during the April 2008 survey activities (Figure 19). ORISE collected three judgmental soil samples from the Pump Alley excavation during the May 2008 survey activities (Figure 15).

SAMPLE ANALYSIS AND DATA INTERPRETATION

Radiological data and sample media were returned to the ORISE laboratory in Oak Ridge, Tennessee for analysis and interpretation. Radioassays were performed in accordance with the ORISE Laboratory Procedures Manual (ORISE 2008b). The soil samples were analyzed by gamma spectroscopy for the primary radionuclides-of-concern (ROC), cobalt-60 (Co-60) and cesium-137 (Cs-137). However, spectra were also reviewed for additional gamma-emitting fission and activation products associated with the RSNGS and other identifiable total absorption peaks. The soil sample results were reported in units of picocuries per gram (pCi/g). Smear samples were analyzed for gross alpha and gross beta activity using a low-background gas proportional counter. Smear results and direct measurements for total surface activity were converted to units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2). Embedded piping scan data were reported in units of counts per minute (cpm) to compare with SMUD's gross gamma cpm FSS results.

FINDINGS AND RESULTS

SURFACE SCANS

The scan percent coverage and room area classifications are provided in Table 1.

Auxiliary Building Structural Surfaces

Beta and gamma surface scans determined that localized areas of residual elevated beta and gamma radiation were present on floor and lower wall surfaces within the evaluated survey units (SUs). Residual surface activity levels approaching site-specific DCGLs but less than the SU specific design DCGL elevated measurement comparison ($DCGL_{EMC}$) were limited to small areas that were interspersed throughout the rooms.

Acid Waste and Rad Waste Drain Systems

Gamma scans of the Acid Waste and Rad Waste drains within the evaluated SUs indicated that gamma radiation levels ranged from 400 to 3,700 cpm. For comparison, the CsI(Tl) detector background range for the conduits along the east side of the Turbine Building at the +40 level elevation ranged from 200 to 800 cpm.

A comparison of ORISE and SMUD gamma measurement results for the Acid Waste System and Rad Waste System drain lines indicated elevated gamma radiation levels at approximately the same length/depth and levels as reported by SMUD personnel in the preliminary FSS data packages. The ORISE confirmatory and SMUD gamma measurement ranges are provided in Table 2.

Pump Alley and Fuel Storage Building Excavation Areas

Gamma scans of the remediated portions of the Pump Alley and Fuel Storage Building exterior excavation areas detected residual elevated gamma radiation levels at several locations within the Pump Alley and did not detect any locations of elevated gamma radiation in the Fuel Storage Building excavation area. Beta scans of the Pump Alley concrete walls indicated several locations of elevated residual beta radiation; these locations were marked for further investigation.

SURFACE ACTIVITY LEVELS

Surface activity and removable activity level results are presented in Table 3.

Auxiliary Building Structural Surfaces

Beta measurements were performed at locations of residual elevated beta and gamma radiation detected during surface scans. Total net beta activity measurements ranged from 270 to 120,000 dpm/100 cm^2. Thirteen of the 59 direct measurements exceeded the DCGL but all were less than the specific SU DCGL$_{EMC}$. Removable gross alpha and gross beta activity ranged from 0 to 18 and -4 to 650 dpm/100 cm^2, respectively.

Pump Alley

Beta measurements were performed at locations of residual elevated beta and gamma radiation detected during surface scans. Total net beta activity measurements ranged from 6,300 to 52,000 dpm/100 cm^2. Two of the eleven direct measurements exceeded the DCGL but were less than the specific SU DCGL$_{EMC}$. Removable gross alpha and gross beta activity ranged from 0 to 1 and -2 to 10 dpm/100 cm^2, respectively.

SOIL SAMPLES

The radionuclide concentrations for the two soil samples collected within the remediated portion of the Fuel Storage Building exterior excavated area ranged from -0.01 to 0.00 pCi/g for Co-60 and 0.42 to 0.44 pCi/g for Cs-137. The radionuclide concentrations for the three soil samples collected within the remediated portion of the Pump Alley ranged from 0.03 to 1.41 pCi/g for Co-60 and 0.38 to 121 pCi/g for Cs-137. The confirmatory radionuclide concentrations for the soil samples are provided in Table 4.

COMPARISON OF SURVEY RESULTS WITH GUIDELINES

STRUCTURAL SURFACE ACTIVITY LEVELS

The major contaminants identified by SMUD at RSNGS are beta-gamma emitters-fission and activation products-resulting from reactor operation. Cs-137 and Co-60 have been identified during characterization as the predominant radionuclides present on structural surfaces. SMUD developed site-specific DCGLs, which were recently approved by the NRC, based on a dose modeling to future occupants not to exceed 25 mrem/year total effective dose equivalent (TEDE) as presented in Section 6 of the LTP (SMUD 2006a and NRC 2007). The DCGLs for surfaces were modified by SMUD to reflect the ratio of radionuclide concentrations (account for the presence of unmeasured contaminants based on contaminant ratios) in the specific SUs that were being evaluated. The applicable surface activity guidelines for the evaluated structural surfaces for these surveys are provided in Table 5. These DCGLs were provided in the preliminary FSS data packages for each evaluated SU and were derived from the LTP and decommissioning technical basis document (DTBD)-05-015 (SMUD 2006a and b).

Confirmatory survey data for Auxiliary Building structural surfaces were compared with the site-specific DCGL for the evaluated Auxiliary Building SUs. Of the 70 direct beta activity measurement results on the concrete structural surfaces, 15 exceeded the Gross Beta DCGL of 43,000 dpm/100 cm^2. Using the gross activity DCGL as determined in DTBD-05-015 (SMUD 2006b) and the area factor determined for each SU, SMUD calculated Design DCGL$_{EMC}$

values which are also provided in Table 5. All confirmatory direct surface activity measurements on the Auxiliary Building structural surfaces in the evaluated SUs were within the site-specific SU $DCGL_{EMC}$ as provided by SMUD in the preliminary FSS data packages.

ACID WASTE AND RAD WASTE DRAIN SYSTEMS

Co-60 is the primary ROC within the embedded piping. SMUD has established a dose-based restriction for embedded piping not to exceed 25 mrem/year that assumes a building occupancy scenario within rooms where embedded piping is present. The corresponding modeled DCGL is 100,000 dpm/100 cm^2. SMUD's grouting action level for embedded piping is 21,000 dpm/100 cm^2 (SMUD 2007).

ORISE's confirmatory drain line results were not directly compared to the embedded piping DCGL; instead, since ORISE and SMUD used similar CsI(Tl) detectors (Ludlum Model 44-159), ORISE compared gross gamma scan readings with either SMUD's preliminary FSS data package gamma scan results for each surveyed pipe at various depths or with background levels as determined during a previous ORISE confirmatory survey (ORISE 2007b).

Confirmatory survey data for the Acid Waste and Rad Waste System drain lines were compared with the preliminary FSS data package gross gamma cpm results. The confirmatory gamma scan results indicated that ORISE gross gamma radiation levels within the drain line pipes were consistent with the SMUD preliminary FSS data package results.

SOIL SAMPLES

Table 6-5 (Table 6) from the LTP provides the single nuclide DCGLs for soil at RSNGS. The DCGL is 12.6 pCi/g for Co-60 and 52.8 pCi/g for Cs-137 (SMUD 2006a). The Fuel Storage Building exterior excavation area soil sample concentrations were below the respective single radionuclide DCGLs. Two of the three confirmatory soil samples from the Pump Alley were well below the single nuclide DCGL; however, one soil sample (1695S0016 containing 121 pCi/g of Cs-137) exceeded the soil DCGL for Cs-137 but was within the soil $DCGL_{EMC}$ for the SU which was calculated by SMUD to be 189 pCi/g.

SUMMARY

During the period of April 27 through 29 and May 29, 2008, ORISE performed confirmatory radiological survey activities which included beta and gamma surface scans, beta activity direct measurements, and removable gross alpha and gross beta activity measurements on structural surfaces within the Auxiliary Building; gamma scans within Auxiliary Building Acid Waste and Rad Waste drains; and gamma scans and the collection of soil samples from the Pump Alley and Fuel Storage Building excavation areas.

Beta and gamma surface scans identified several areas of elevated beta radiation on the structural surfaces of the evaluated SUs within the Auxiliary Building. Additional investigations of these locations indicated that the majority of the elevated radiation levels were attributable to localized areas of residual beta-gamma radiation within the matrix of the concrete media. In general, the elevated surface activity was limited to small areas that were interspersed throughout the rooms.

Direct measurements were performed at 70 locations. Fifteen direct measurements exceeded the site-specific gross beta DCGL but all were within the SU specific design $DCGL_{EMC}$ criteria. The confirmatory survey results for Rooms 10, 15, 40, 42, 51, 52 and the Pump Alley concrete walls are in agreement with the radiological status of these SUs as presented in the licensee's preliminary FSS data packages.

Confirmatory gamma surface scans of the evaluated Auxiliary Building drain systems indicated that the gamma radiation levels were consistent with the results presented in the preliminary FSS data packages for the surveyed drains.

The soil sample results from the Fuel Storage Building exterior excavation area were below the individual radionuclide DCGLs and meet the soil release criteria. One of the three soil samples from the Pump Alley exceeded the soil DCGL for Cs-137; however, this sample was within the SU specific design $DCGL_{EMC}$ for Cs-137.

FIGURES

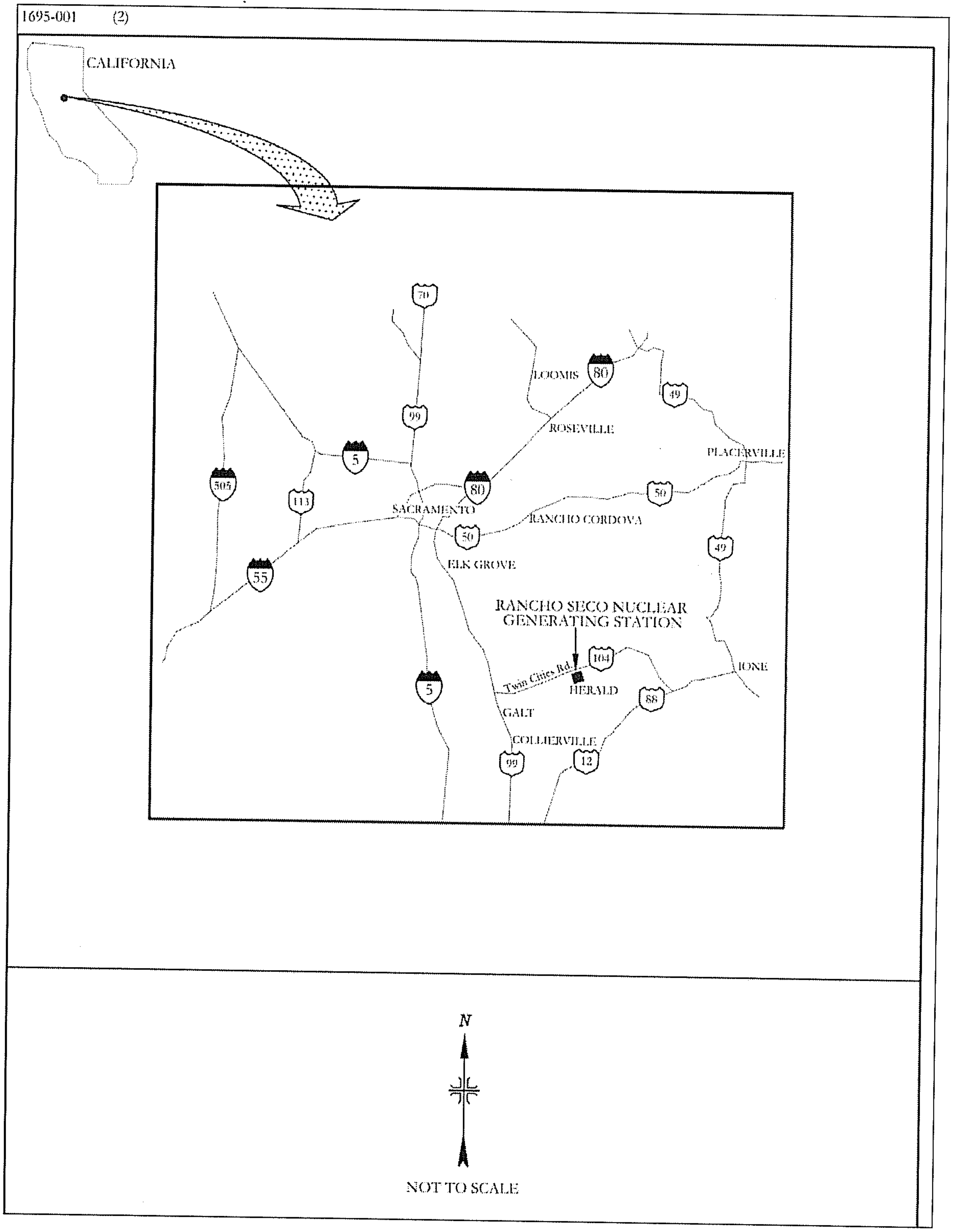


Figure 1: Location of Rancho Seco Nuclear Generating Station, Herald, California

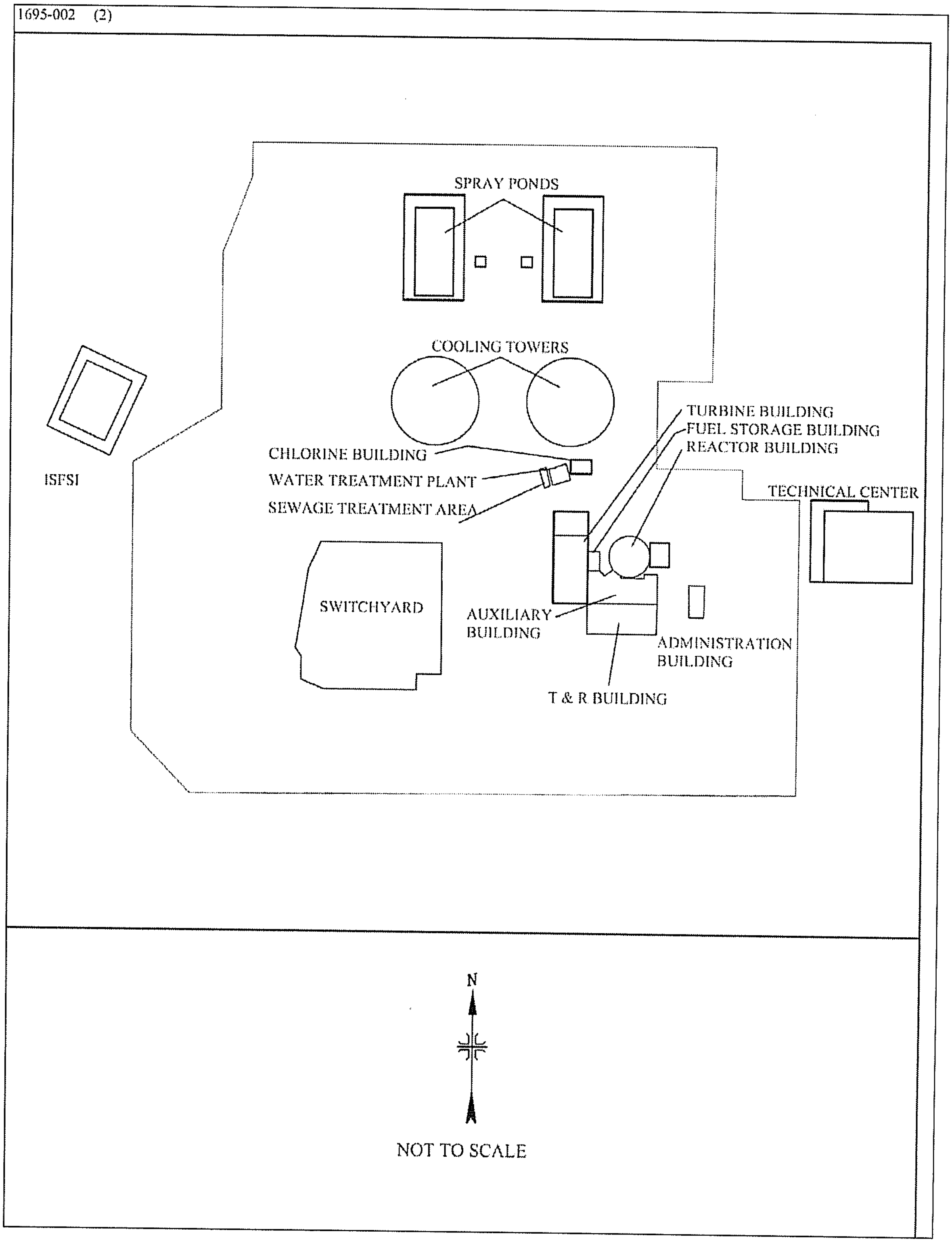
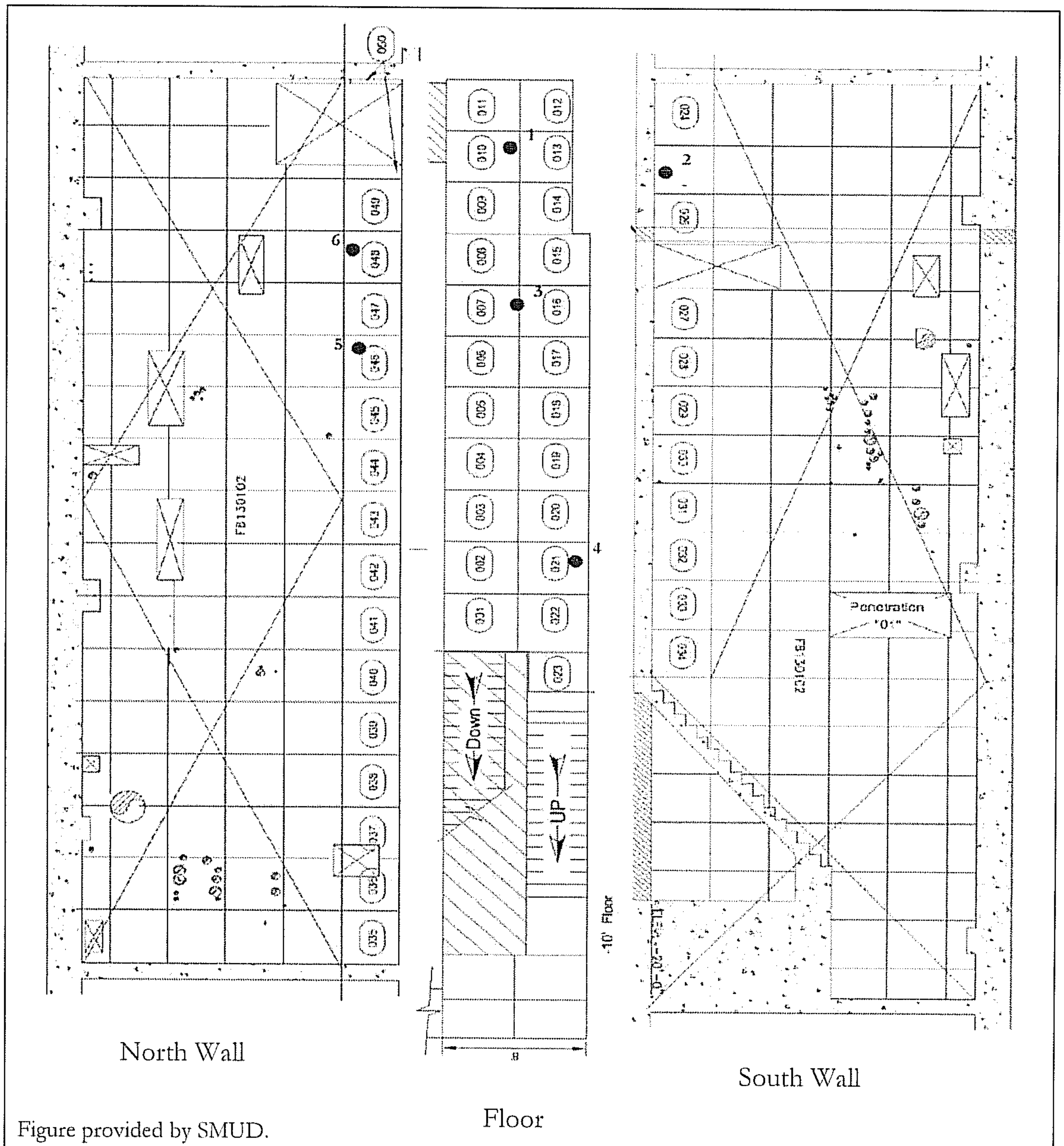


Figure 2: Plot Plan of the Industrial Area at Rancho Seco Nuclear Generating Station



MEASUREMENT/SAMPLE LOCATION	
● # Single Point, Floor and Lower Walls	N ◄ NOT TO SCALE

Figure 3: Survey Unit F8130101, Room 10 - Direct Measurement and Sample Locations

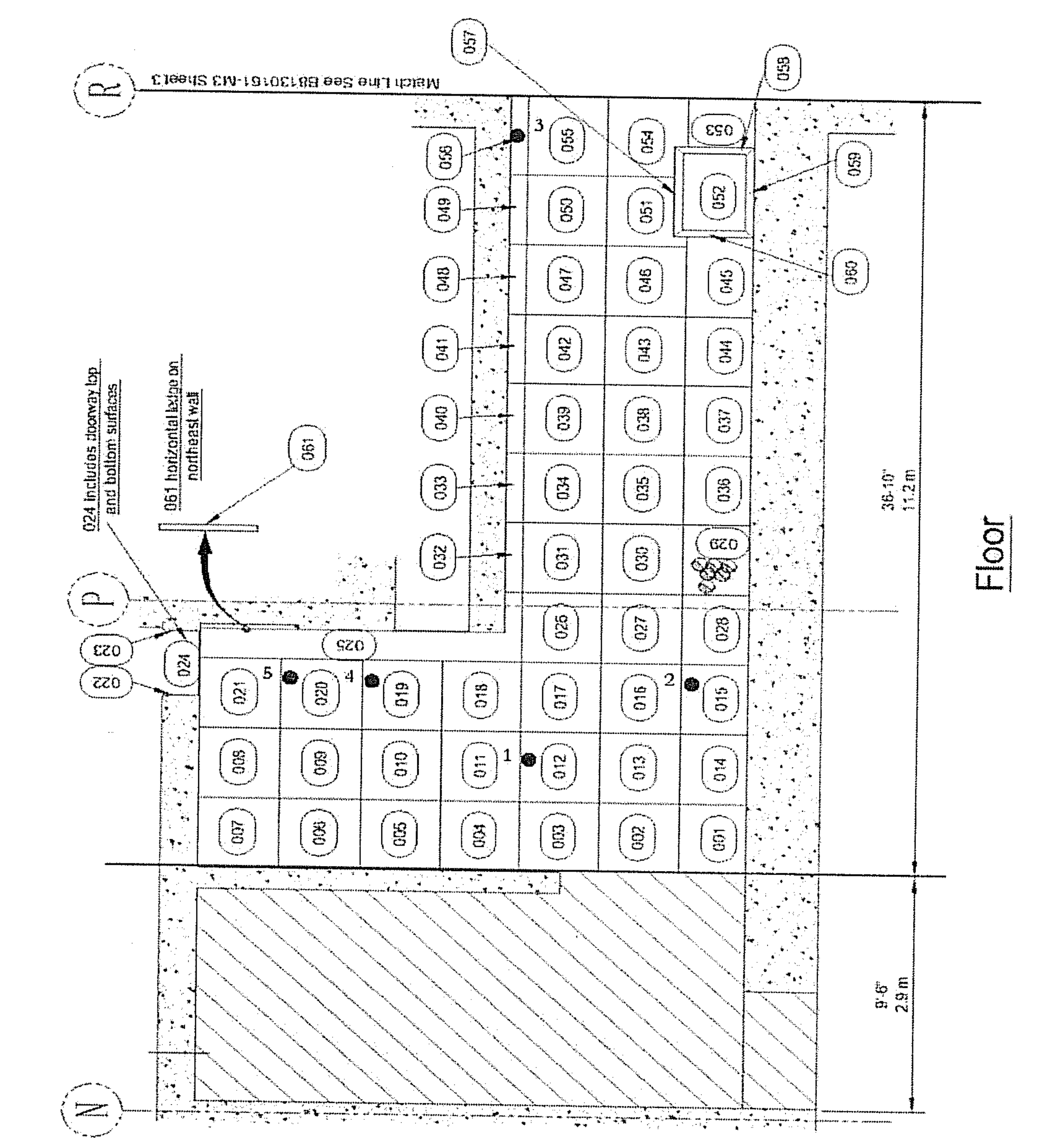


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	N
●# Single Point, Floor and Lower Walls	NOT TO SCALE

Figure 4: Survey Unit F8130151, Room 15 - Direct Measurement and Sample Locations

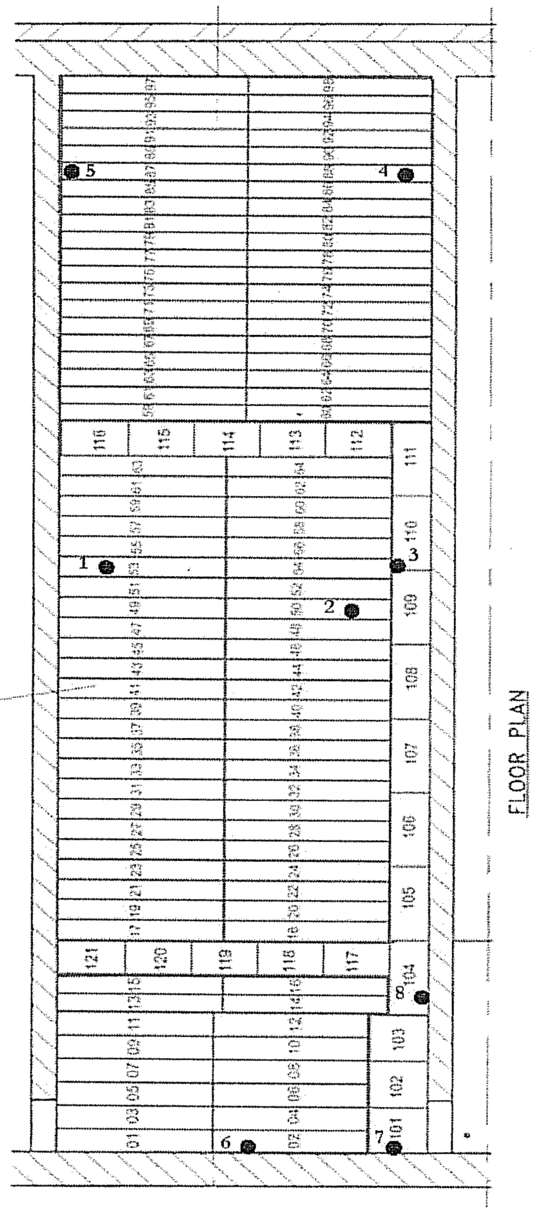


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	N ←
●# Single Point, Floor and Lower Walls	NOT TO SCALE

Figure 5: Survey Unit F8130541, Room 40 - Direct Measurement and Sample Locations

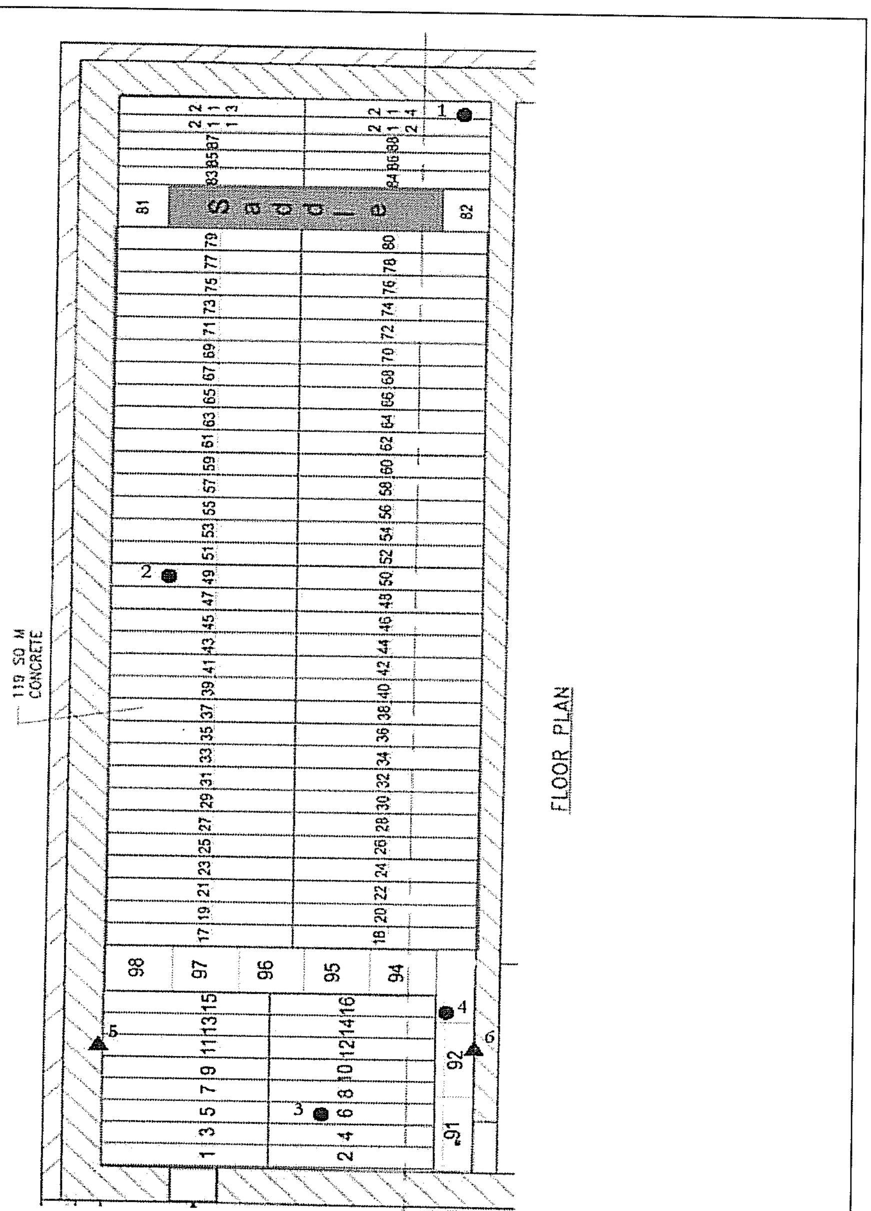


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	
● # Single Point, Floor and Lower Walls ▲ # Single Point, Upper Surfaces	N ◄ NOT TO SCALE

Figure 6: Survey Unit F8130581, Room 42 - Direct Measurement and Sample Locations

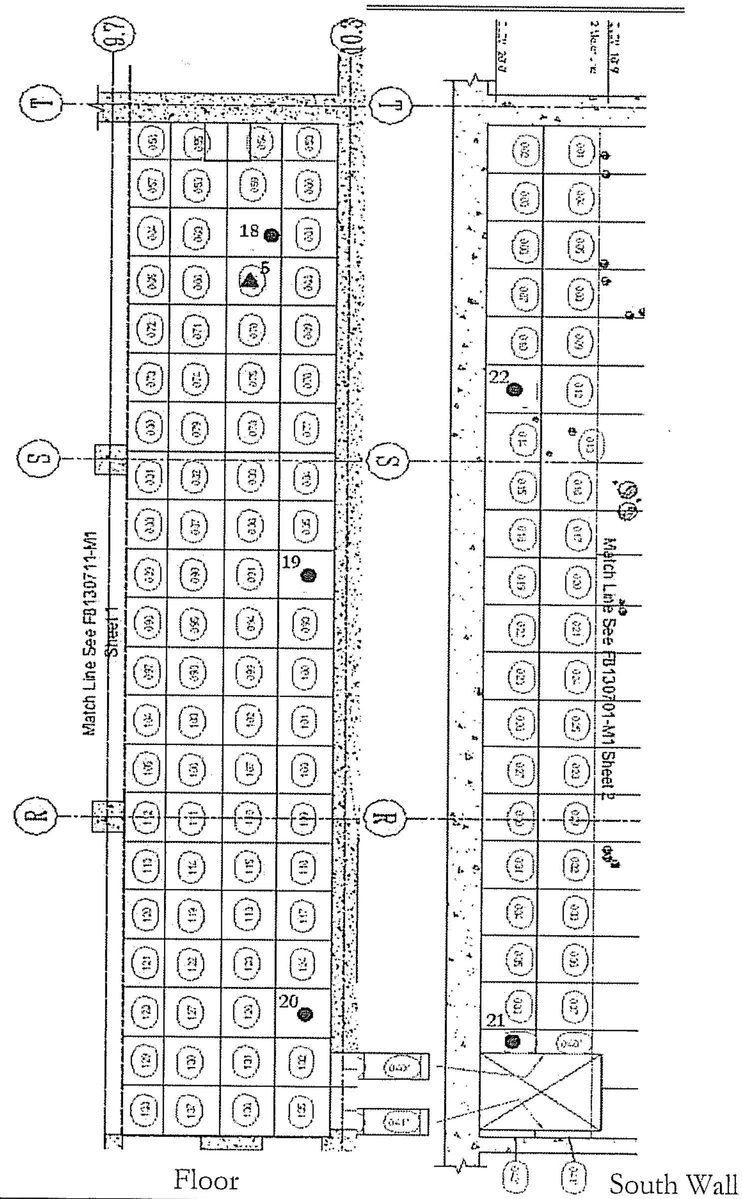


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	
● # Single Point, Floor and Lower Walls ▲ # Single Point, Upper Surfaces	N ◄ NOT TO SCALE

Figure 7: Survey Unit F8130169, F8130701 and F8130711, Room 51, Section 1 - Direct Measurement and Sample Locations

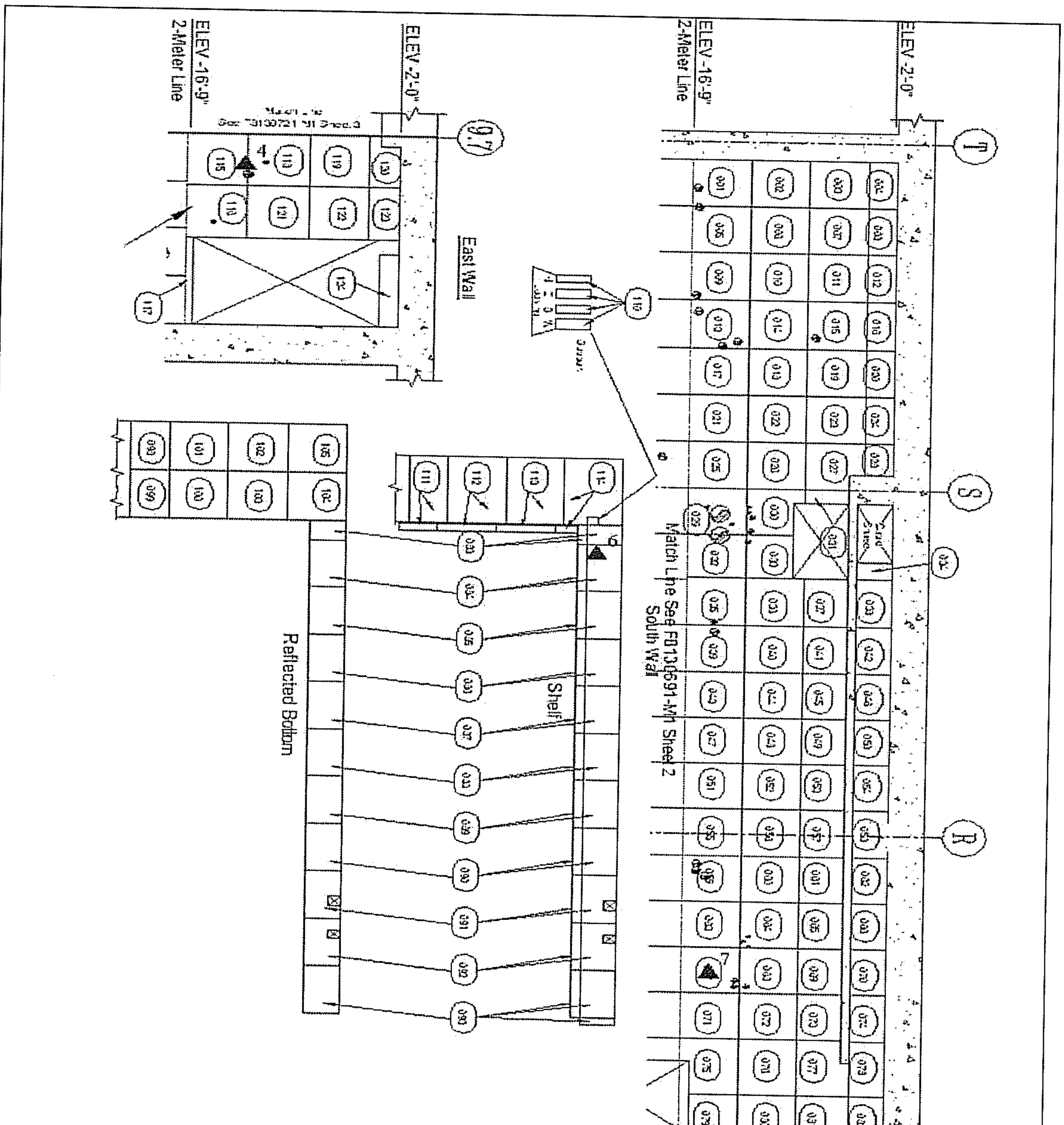
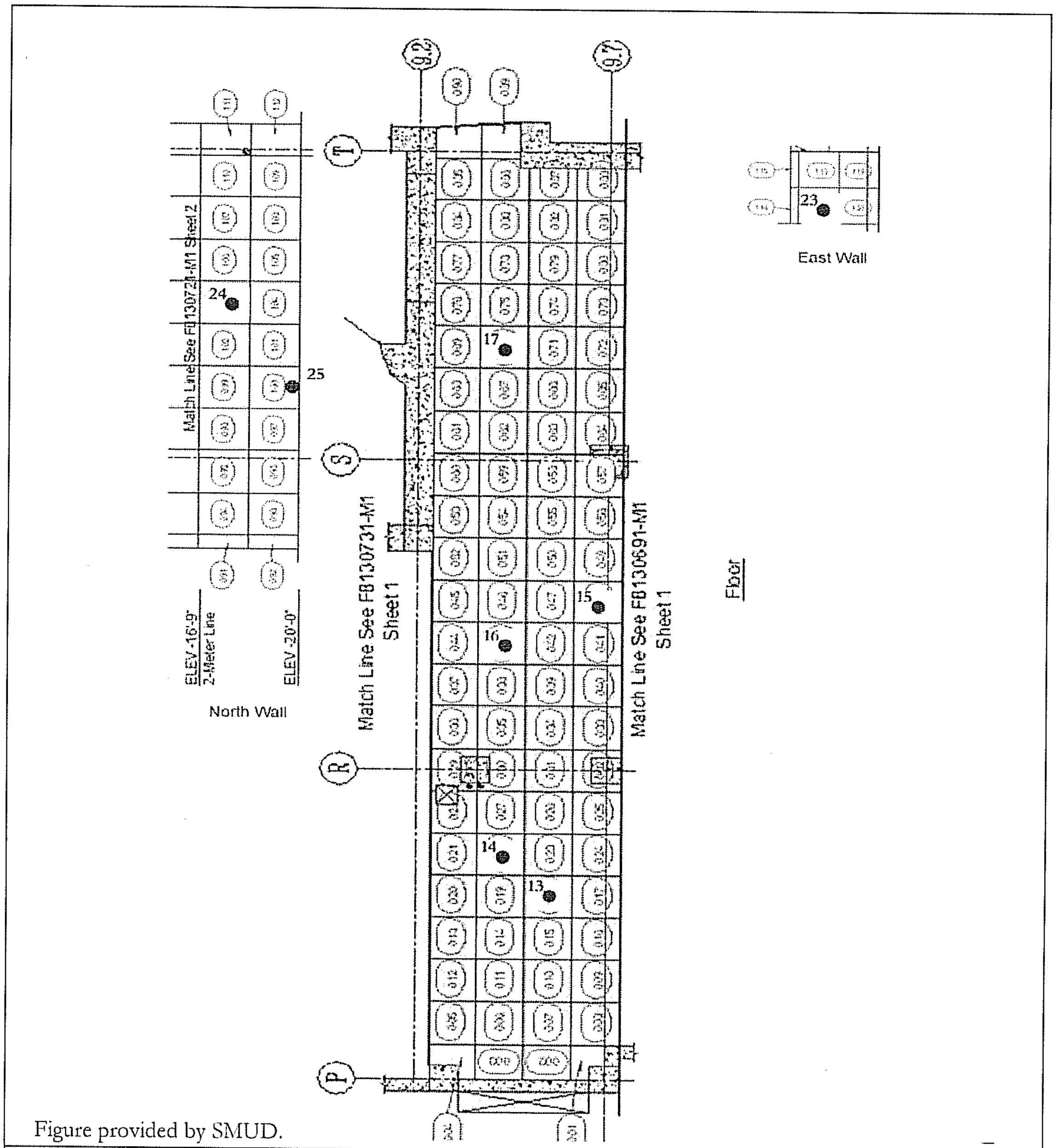


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION ▲ # Single Point, Upper Surfaces	N NOT TO SCALE

Figure 8: Survey Unit F8130701, Room 51, Section 1, Upper Surfaces - Direct Measurement and Sample Locations



MEASUREMENT/SAMPLE LOCATION

●# Single Point, Floor and Lower Walls

N

NOT TO SCALE

Figure 9: Survey Unit F8130701 and F8130711, Room 51, Section 2 - Direct Measurement and Sample Locations

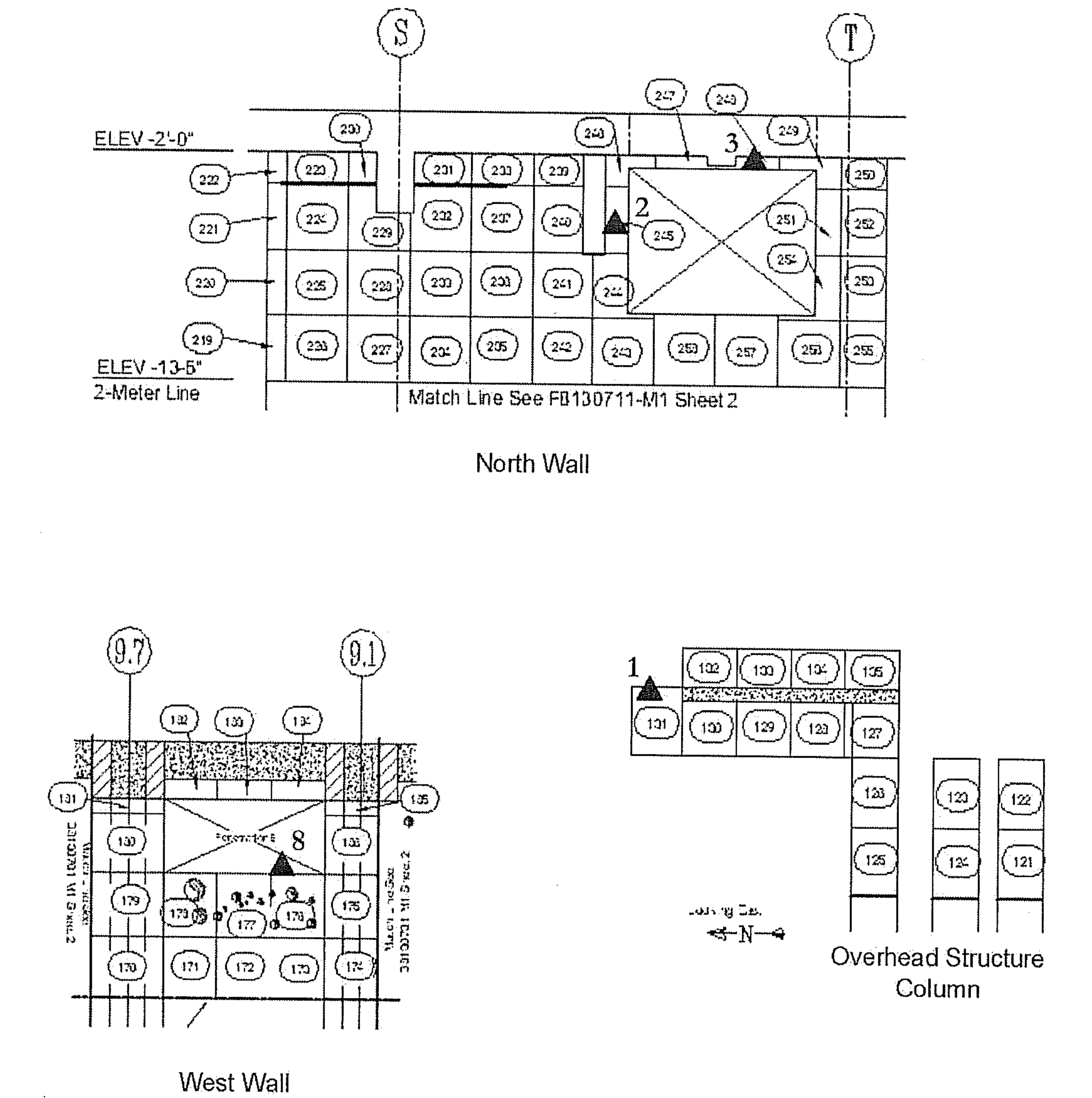


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	
▲ # Single Point, Upper Surfaces	N ← NOT TO SCALE

Figure 10: Survey Unit F8130721, Room 51, Section 2 - Direct Measurement and Sample Locations

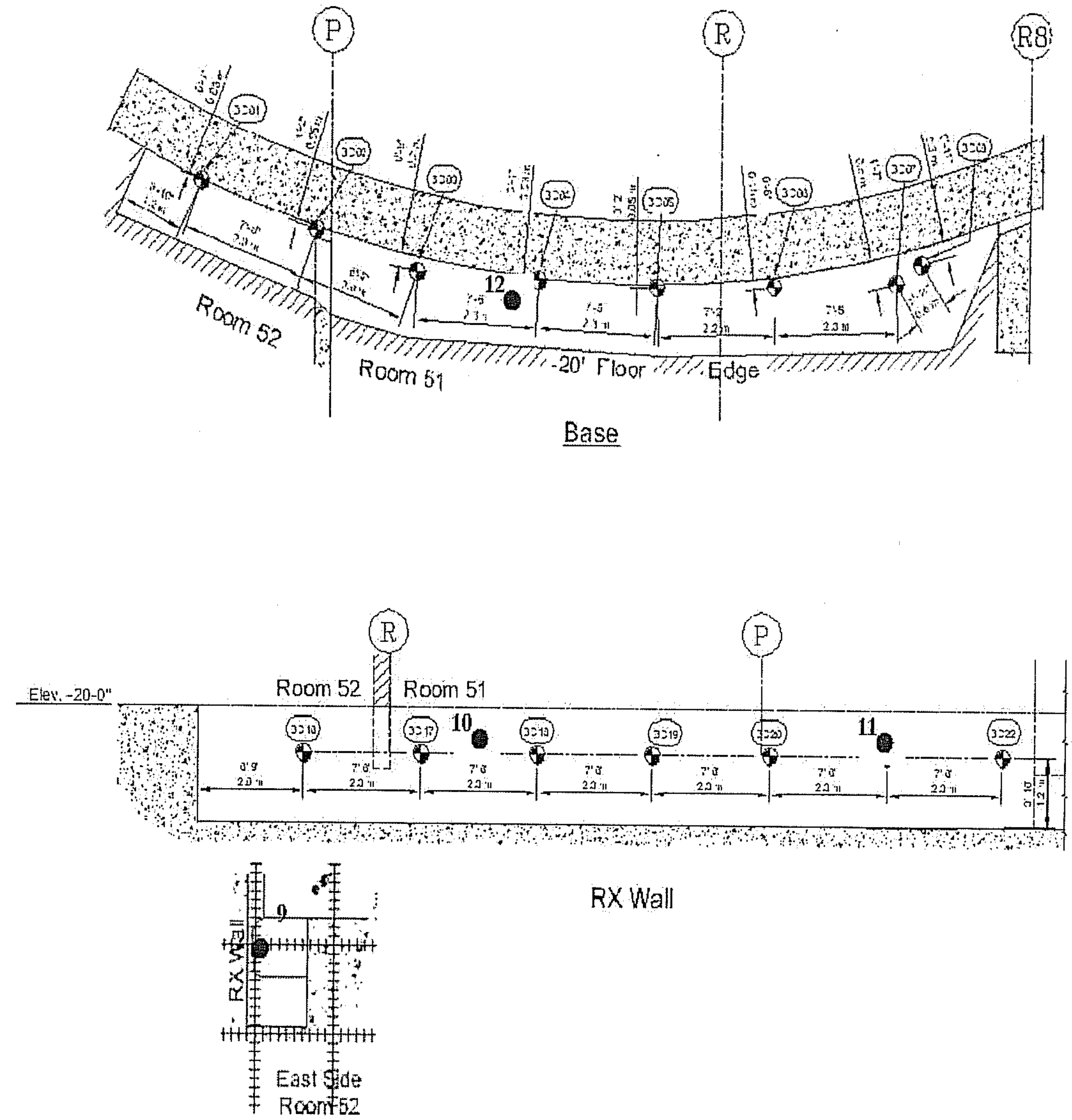


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	N
●# Single Point, Floor and Lower Walls	NOT TO SCALE

Figure 11: Survey Unit F8130732, Room 51 and 52 Excavation - Direct Measurement and Sample Locations

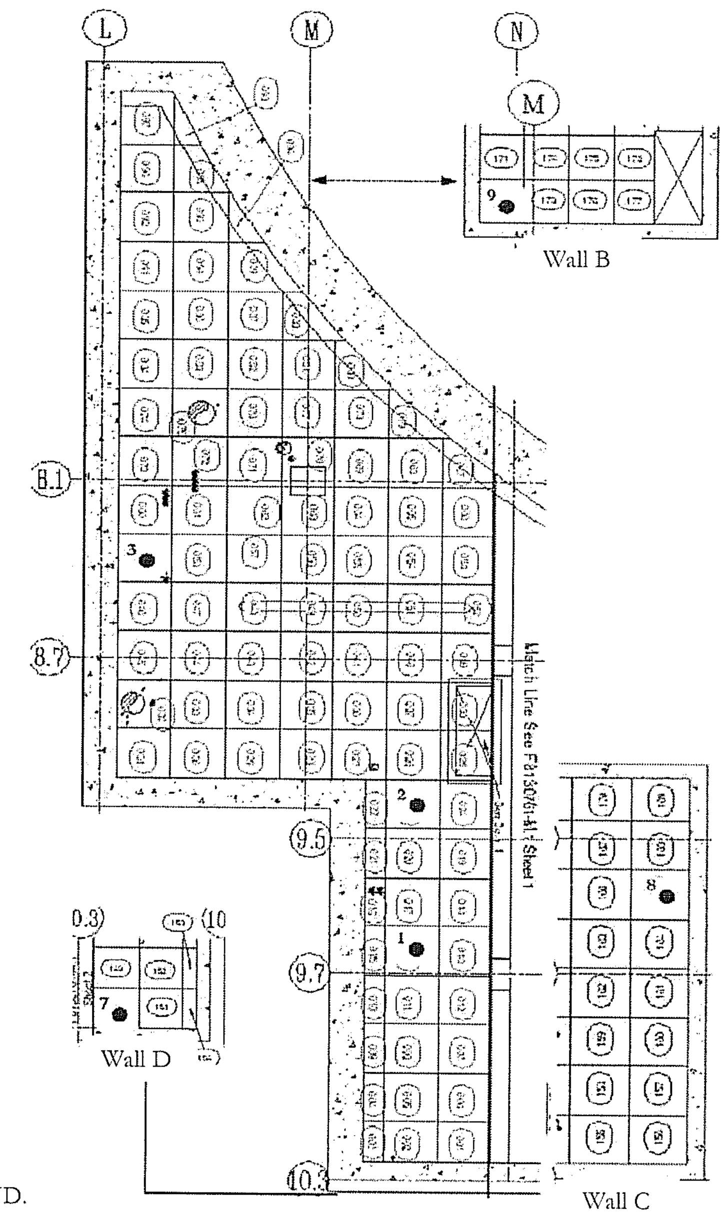


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	N
●# Single Point, Floor and Lower Walls	NOT TO SCALE

Figure 12: Survey Unit F8130751, Room 52, West of Column N - Direct Measurement and Sample Locations

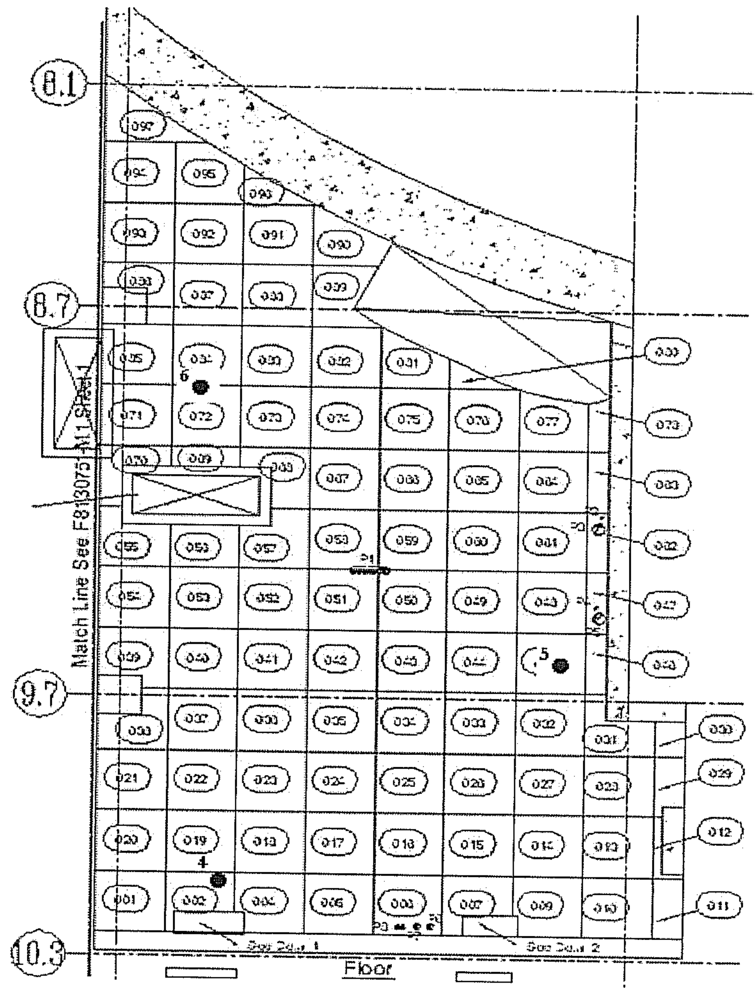


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION

●# Single Point, Floor and Lower Walls

N

NOT TO SCALE

Figure 13: Survey Unit F8130761, Room 52, East of Column N - Direct Measurement and Sample Locations

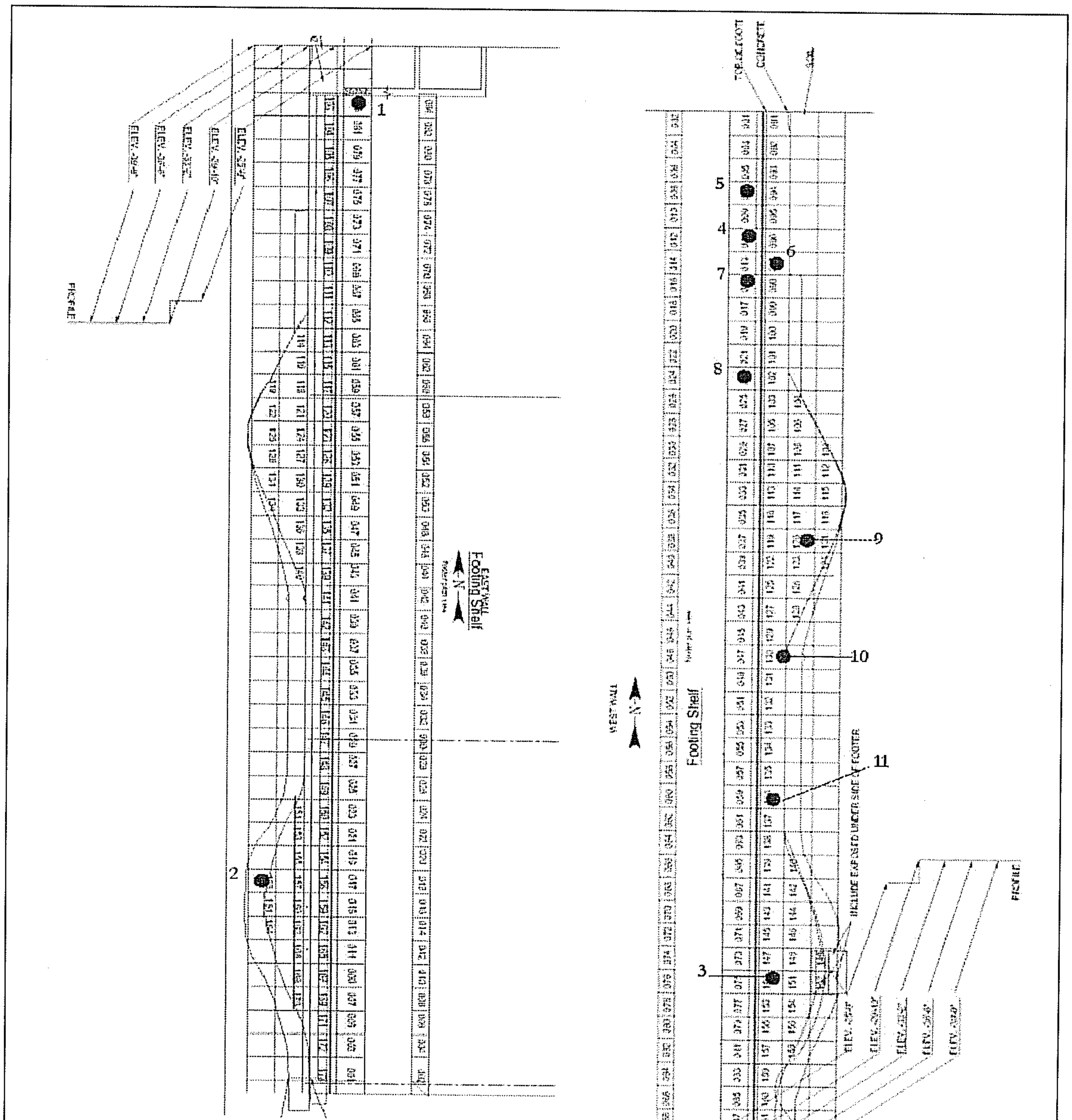


Figure provided by SMUD.

MEASUREMENT/SAMPLE LOCATION	
● # Single Point, Floor and Lower Walls	N NOT TO SCALE

Figure 14: Survey Unit F8130461, Pump Alley - Direct Measurement and Sample Locations

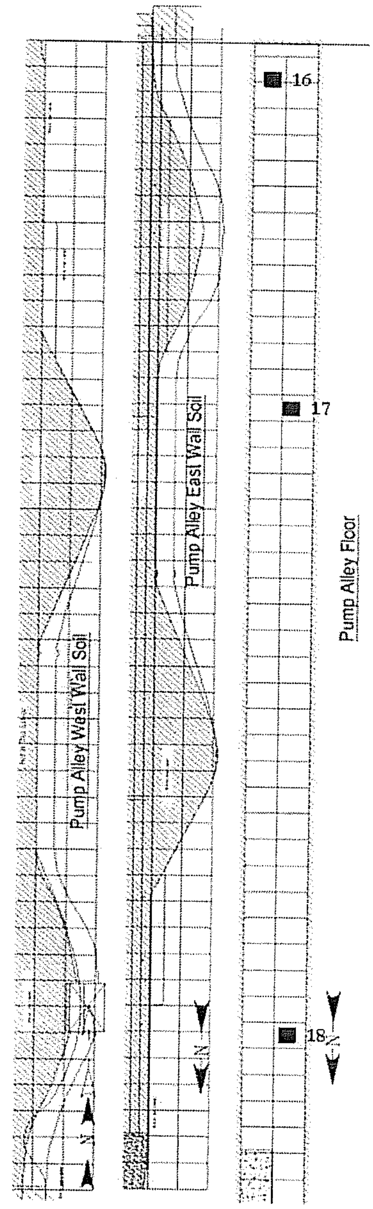


Figure provided by SMUD.

SOIL SAMPLE LOCATION ■ # Soil Sample	NOT TO SCALE

Figure 15: Survey Unit F8130421, Pump Alley – Soil Sample Locations

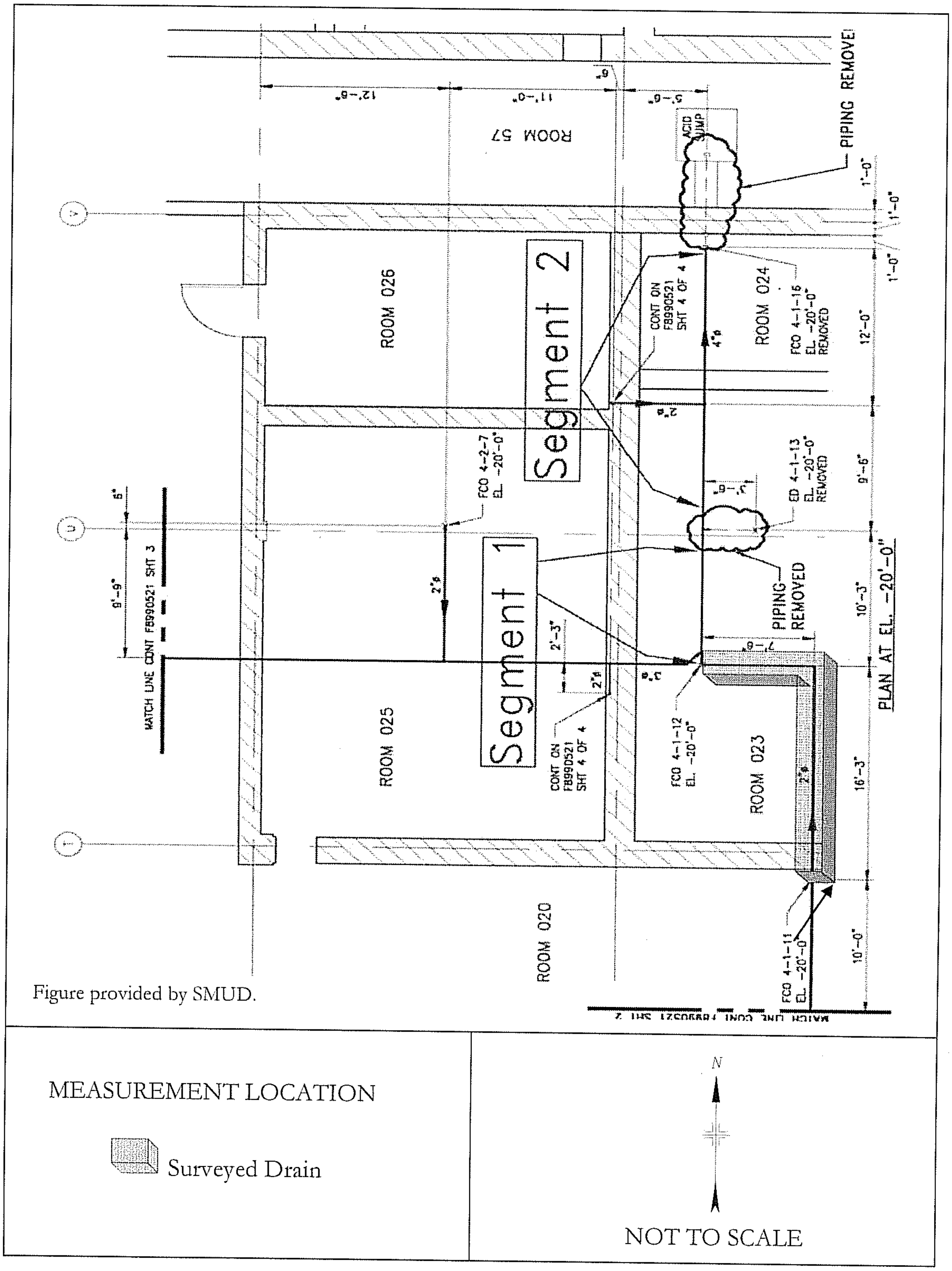


Figure provided by SMUD.

Figure 16: Survey Unit F8990521, Acid Waste Drains – Surveyed Drain 4-1-11

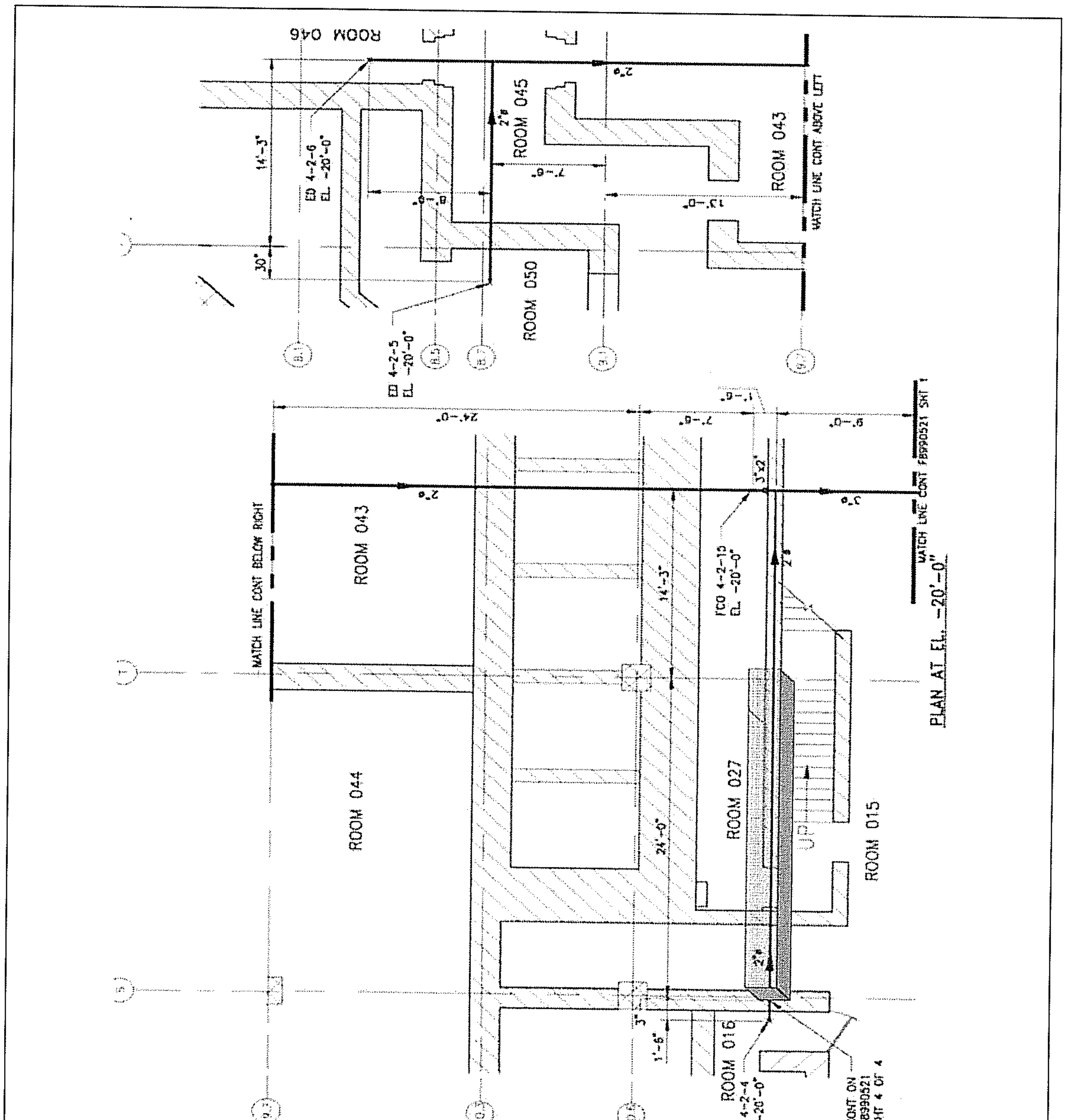


Figure provided by SMUD.

MEASUREMENT LOCATION

Surveyed Drain

N

NOT TO SCALE

Figure 17: Survey Unit F8990521, Acid Waste Drains – Surveyed Drain 4-2-4

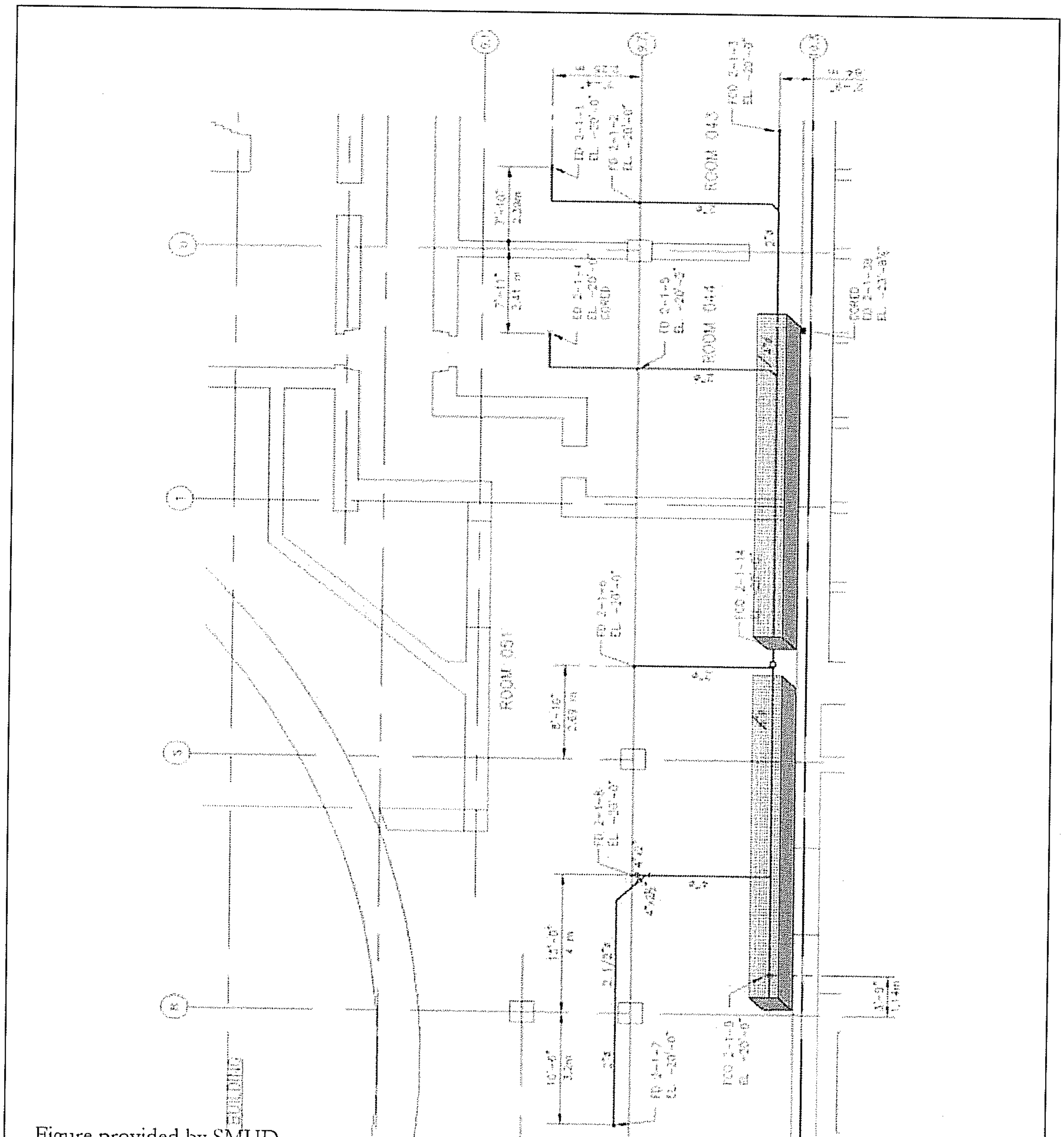


Figure provided by SMUD.

MEASUREMENT LOCATION

Surveyed Drain

N

NOT TO SCALE

Figure 18: Survey Unit F8990521, Rad Waste Drains – Surveyed Drains 2-1-9 and 2-1-14

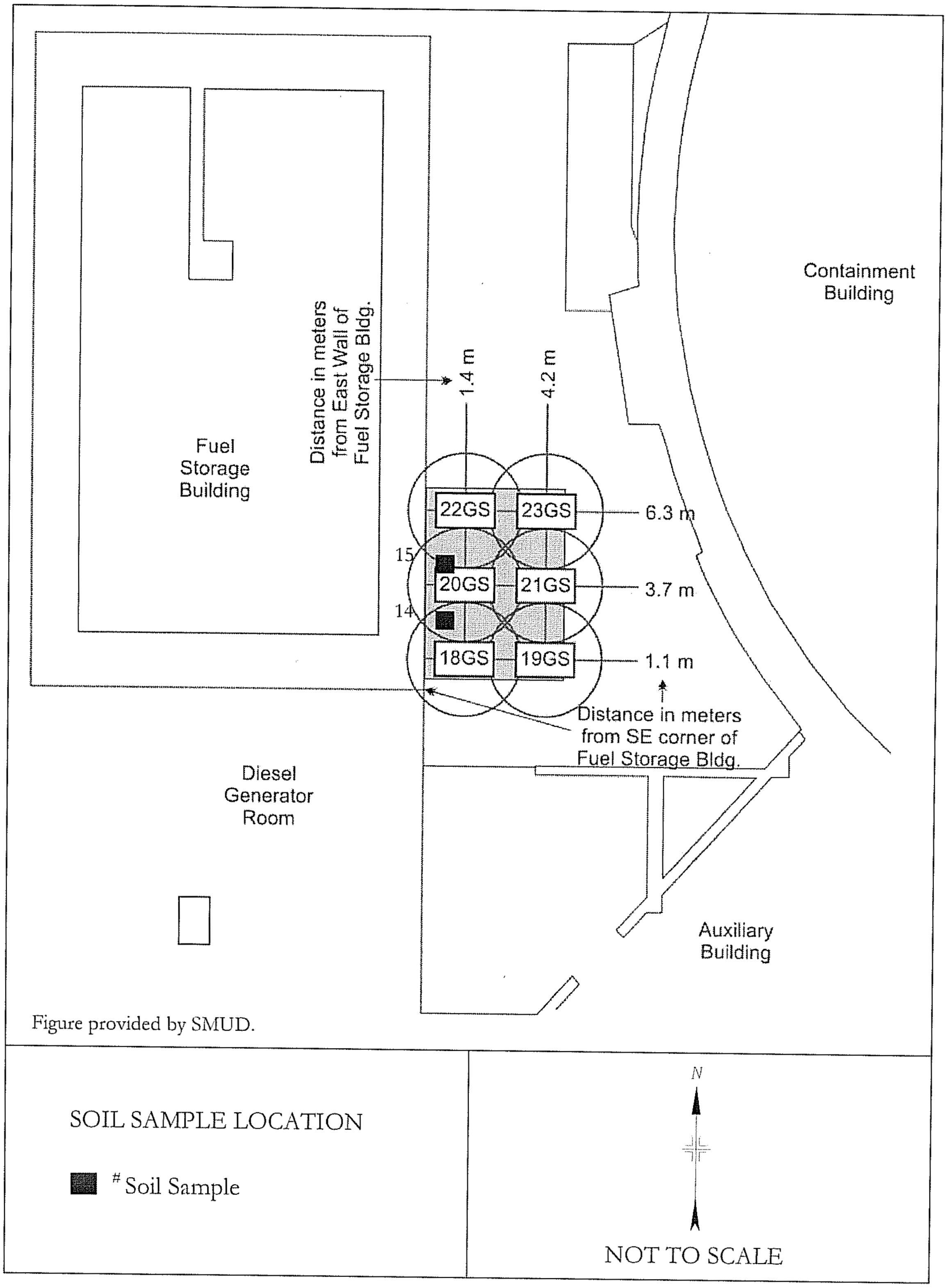


Figure 19: Survey Unit F8130101, Fuel Storage Building Exterior Excavation – Soil Sampling Locations

TABLES

TABLE 1: SURVEY UNIT CLASSIFICATION AND SCAN COVERAGE					
Auxiliary Building Survey Unit/Room[a]	Class	Percent Scan Coverage			
		Gamma Floor/Lower Wall	Beta Floor	Beta Lower Wall	Beta Upper Surfaces
10 FL and LW	1	100/50	100	50	---[b]
15 FL	1	100/50	100	60	---
40 FL and LW	1	100	25	75	---
42 FL and LW	1	100	85	50	---
51 FL and LW	1	50	50	50	---
51 US	1	---	---	---	5
52 FL and LW	1	75	50	50	---
51 and 52 Excavation	1	---	70	70	---
Pump Alley	1	100[c]	---	60	---
Fuel Building Storage Excavation	1	100[c]	---	---	---

[a]Refer to Figures 3 through 15 and 19. FL = floor, LW = lower wall and US = upper surfaces.
[b]Scans not performed.
[c]Gamma surface soil scans of excavations.

TABLE 2: ACID WASTE AND RAD WASTE DRAIN SYSTEM GAMMA MEASUREMENTS				
Drain Line Location	**Diameter (inches)**	**Scan Length (feet)**	**Gamma Scan Range (cpm)**	
			ORISE	**SMUD**
Turbine Building Backgrounds[a]				
Conduit, East Side 1	4	1	300 to 600	---[b]
Conduit, East Side 2	4	1	300 to 600	---
Conduit, East Side 3	4	1	200 to 600	---
Conduit, East Side 4	4	1	300 to 600	---
Penetration, East Side	4	1	300 to 600	---
Exciter Pad East	4	12	200 to 800	---
Exciter Pad West	4	12	200 to 800	---
Background Range	---	---	200 to 800	---
			Gamma Measurements (cpm)[c]	
Auxiliary Building Acid Waste Drains[d]				
4-1-11	2	0	400	450
		1	400	510
		2	500	510
		3	400	500
		4	500	520
		5	500	520
		6	400	550
		7	400	460
		8	500	510
		9	500	510
		10	500	560
		11	500	540
		12	400	450
		13	400	530
		14	500	520
		15	400	510
		16	500	550
		17	400	480
		18	400	510
		19	500	540
		20	500	540
		21	400	510

TABLE 2: ACID WASTE AND RAD WASTE DRAIN SYSTEM GAMMA MEASUREMENTS				
Drain Line Location	Diameter (inches)	Scan Length (feet)	Gamma Measurements (cpm)[c]	
			ORISE	SMUD
Auxiliary Building Acid Waste Drains[d]				
4-2-4	2	0	2,900	2,500
		1	400	370
		2	400	410
		3	500	400
		4	400	440
		5	500	410
		6	500	420
		7	400	350
		8	500	410
		9	500	420
		10	500	380
		11	500	390
		12	400	330
		13	500	400
		14	500	360
		15	400	400
		16	400	390
		17	400	400
		18	400	390
		19	400	410
		20	400	410
		21	500	410
		22	400	370
Auxiliary Building Rad Waste Drains[d]				
2-1-9	6	0	3,100	3,600
		1	1,100	2,300
		2	700	1,100
		3	500	800
		4	500	1,000
		5	900	1,400
		6	2,400	2,900

TABLE 2: ACID WASTE AND RAD WASTE DRAIN SYSTEM GAMMA MEASUREMENTS				
Drain Line Location	Diameter (inches)	Scan Length (feet)	Gamma Measurements (cpm)[c]	
			ORISE	SMUD
Auxiliary Building Rad Waste Drains[d]				
2-1-9 (continued)	6	7	1,900	2,400
		8	1,100	1,900
		9	700	1,400
		10	1,000	1,400
		11	800	1,300
		12	800	1,200
		13	800	1,300
		14	1,000	1,400
		15	1,000	1,500
		16	1,000	1,500
		17	1,000	1,600
		18	1,000	1,600
		19	1,100	1,500
		20	1,100	1,500
2-1-14	4	0	800	1,200
		1	700	1,000
		2	700	1,000
		3	1,700	2,300
		4	1,300	1,700
		5	1,400	1,900
		6	1,500	1,800
		7	700	1,100
		8	800	1,000
		9	1,000	1,100
		10	1,600	2,000
		11	1,600	2,200
		12	2,900	3,700
		13	2,100	3,000
		14	3,700	3,700
		15	2,600	3,000
		16	2,700	2,900

TABLE 2: ACID WASTE AND RAD WASTE DRAIN SYSTEM GAMMA MEASUREMENTS				
Drain Line Location	Diameter (inches)	Scan Length (feet)	Gamma Measurements (cpm)[c]	
			ORISE	SMUD
Auxiliary Building Rad Waste Drains[d]				
2-1-14 (continued)	4	17	2,500	3,000
		18	2,300	2,200
		19	2,600	3,000
		20	2,800	2,900

[a]Turbine Building embedded piping backgrounds were determined within Turbine Building conduits. This data was collected during a previous ORISE survey (ORISE 2007b).
[b]Measurements not performed by SMUD within the Turbine Building conduits.
[c]SMUD gamma scan measurement data was provided to ORISE in preliminary FSS data packages or during comparison gamma measurements made with similar gamma detectors. ORISE and SMUD results were rounded to two significant digits.
[d]Refer to Figures 16 through 18.

TABLE 3: SURFACE ACTIVITY LEVELS					
Room/ Location[a]	Surface[b]	Total Beta Activity (dpm/100 cm^2)[c]	Removable Activity (dpm/100 cm^2)		Activity Meets Gross Beta DCGL[d]
			Alpha	Beta	
Auxiliary Building, Room 10					
1	FL	9,800	0	16	YES
2	LW	6,500	0	8	YES
3	FL	7,400	0	8	YES
4	FL	31,000	1	84	YES
5	LW	7,800	0	46	YES
6	LW	9,900	5	190	YES
Auxiliary Building, Room 15					
1	FL	8,700	0	76	YES
2	FL	11,000	1	2	YES
3	FL	11,000	1	6	YES
4	FL	21,000	0	2	YES
5	FL	42,000	1	1	YES
Auxiliary Building, Room 40					
1	FL	37,000	0	-4	YES
2	FL	20,000	3	-3	YES
3	TW	14,000	0	-1	YES
4	FL	16,000	0	1	YES
5	LW	30,000	0	14	YES
6	LW	16,000	0	1	YES
7	LW	34,000	0	1	YES
8	LW	27,000	0	-1	YES
Auxiliary Building, Room 42					
1	FL	53,000	5	110	NO
2	FL	9,500	1	34	YES
3	FL	12,000	0	2	YES
4	FL	8,000	0	-3	YES
5	US	6,600	0	36	YES
6	US	18,000	1	12	YES

TABLE 3: SURFACE ACTIVITY LEVELS					
Room/ Location[a]	Surface[b]	Total Beta Activity (dpm/100 cm^2)[c]	Removable Activity (dpm/100 cm^2)		Activity Meets Gross Beta DCGL[d]
			Alpha	Beta	
Auxiliary Building, Room 51					
1	US	75,000	0	5	NO
2	US	270	1	4	YES
3	US	11,000	9	63	YES
4	US	3,400	0	7	YES
5	US	510	0	1	YES
6	US	4,300	0	6	YES
7	US	75,000	13	210	NO
8	US	5,500	0	3	YES
9	US	120,000	0	46	NO
10	US	31,000	0	10	YES
11	US	55,000	3	140	NO
12	US	4,500	0	-4	YES
13	FL	28,000	0	3	YES
14	FL	18,000	3	-2	YES
15	FL	63,000	0	2	NO
16	FL	9,200	1	3	YES
17	FL	6,500	3	-1	YES
18	FL	32,000	0	6	YES
19	FL	65,000	1	16	NO
20	FL	21,000	0	1	YES
21	LW	27,000	0	1	YES
22	LW	51,000	0	6	NO
23	LW	15,000	18	290	YES
24	LW	55,000	7	650	NO
25	LW	22,000	1	1	YES

TABLE 3: SURFACE ACTIVITY LEVELS					
Room/ Location[a]	Surface[b]	Total Beta Activity (dpm/100 cm^2)[c]	Removable Activity (dpm/100 cm^2)		Activity Meets Gross Beta DCGL[d]
			Alpha	Beta	
Auxiliary Building, Room 52					
1	FL	56,000	0	14	NO
2	FL	10,000	0	-2	YES
3	FL	54,000	0	-1	NO
4	FL	35,000	0	3	YES
5	FL	13,000	3	-1	YES
6	FL	27,000	0	-3	YES
7	LW	61,000	5	200	NO
8	LW	64,000	0	190	NO
9	LW	14,000	0	18	YES
Auxiliary Building, Pump Alley					
1	LW	31,000	0	10	YES
2	LW	31,000	0	8	YES
3	LW	6,300	1	-2	YES
4	LW	52,000	0	7	NO
5	LW	45,000	0	1	NO
6	LW	10,000	1	1	YES
7	LW	11,000	0	4	YES
8	LW	12,000	0	2	YES
9	LW	13,000	0	1	YES
10	LW	10,000	0	10	YES
11	LW	15,000	0	8	YES

[a]Refer to Figures 3 through 14.
[b]Structural surfaces; FL = floor, LW = lower wall, TW = trench wall and US = upper surfaces.
[c]Direct measurement results rounded to two significant digits.
[d]DCGL values are provided in Table 5. All surface activity measurements that were greater than the gross beta DCGL were less than the design DCGLs determined for each specific survey unit.

TABLE 4: RADIONUCLIDE CONCENTRATIONS IN SOIL SAMPLES (pCi/g)		
Sample Location[a]	Co-60	Cs-137
Fuel Storage Building Excavation		
14	-0.01 ± 0.04[b]	0.44 ± 0.05
15	0.00[c] ± 0.04	0.42 ± 0.05
Pump Alley		
16	1.41 ± 0.13	121 ± 11
17	0.22 ± 0.04	15.7 ± 1.4
18	0.03 ± 0.06	0.38 ± 0.06

[a]Refer to Figures 15 and 19.
[b]Uncertainties represent the 95% confidence level based on total propagated uncertainties.
[c]Zero values are due to rounding.

TABLE 5: DERIVED CONCENTRATION GUIDELINE LEVELS AND ELEVATED MEASUREMENT COMPARISONS FOR SURVEYED ROOMS			
Auxiliary Building Survey Unit/Room[a]	Class	Gross Beta DCGL[b] (dpm/100 cm^2)	Design $DCGL_{EMC}$[c] (dpm/100 cm^2)
10 FL and LW	1	43,000	224,000
15 FL and LW	1	43,000	152,400
40 FL and LW	1	43,000	141,900
42 FL and LW	1	43,000	141,900
51 FL and LW	1	43,000	154,800
51 US	1	43,000	150,500
52 FL and LW	1	43,000	154,800
51 and 52 Excavation	1	43,000	176,300
Pump Alley LW	1	43,000	153,300

[a]Refer to Figures 3 through 14. FL = floor, LW = lower wall and US = upper surfaces.
[b]Gross beta DCGL accounts for radionuclide fractions and hard to detects as specified in the DTBD-05-15 (SMUD 2006b).
[c]$DCGL_{EMC}$ provided by SMUD and accounted for area factors determined for each specific survey unit.

TABLE 6: SINGLE NUCLIDE $DCGL_W$ VALUES FOR DETECTABLE RADIONUCLIDES[a]		
Radionuclide	**Peak of the Mean Dose (mrem/y per pCi/g)**	**$DCGL_W$ (pCi/g)**
C-14	2.93E-06	8.33E+06
Co-60	1.93E+00	1.26E+01
Ni-63	1.60E-06	1.52E+07
Sr-90	3.76E-03	6.49E+03
Cs-134	1.09E+00	2.24E+01
Cs-137	4.62E-01	5.28E+01

[a]Table 6-5 from the License Termination Plan (SMUD 2006a).

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SUNSI Review Complete: EMG ADAMS: ■Yes □No Initials: EMG
■Publicly Available □Non-Publicly Available □Sensitive ■Non-Sensitive

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