UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

August 31, 2001

Stephen M. Quennoz, Vice President,
Power Supply/Generation
Portland General Electric Company
Trojan Nuclear Plant
71760 Columbia River Highway
Rainier, Oregon 97048

SUBJECT: NRC INSPECTION REPORT 50-344/01-02

Dear Mr. Quennoz:

An NRC inspection was conducted on May 14-17, 2001, at your Trojan Nuclear Plant. On August 7and 22, 2001, additional telephonic discussions were held. The enclosed report presents the scope and results of this inspection and the subsequent phone conversations.

The primary purposes of this inspection were to review the results of your final radiological survey of the interior surface of the reactor building containment dome completed on May 8, 2001, and to conduct an independent NRC confirmatory survey of the containment dome to verify your survey results. This is the first in a series of NRC inspections which will focus on the adequacy of your final survey program to support the eventual decision that your facility has been remediated below regulatory limits specified in 10 CFR 20, Subpart E. No violations of NRC regulations were identified during this inspection.

As a result of this inspection, an issue was identified concerning the survey unit size for the containment dome. This was discussed with your staff during both the August 7 and 22, 2001, phone calls. The survey unit size selected for containment was 1910 m^2. This differs from the recommended size of 1000 m^2 for Class 2 structures specified in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). MARSSIM is considered acceptable guidance for determining survey unit size and provides a recommended upper limit to ensure an adequate number of survey data points are collected. MARSSIM allows larger survey unit sizes if appropriate rationale is developed and documented. For the containment dome survey recently completed, the NRC conducted a confirmatory survey which validated the results of your survey. Therefore, no additional action is needed for the containment dome survey. However, future confirmatory surveys by the NRC will only include representative areas of your facility and will not include all areas of the Trojan facility. Therefore, for future surveys at Trojan using survey unit sizes exceeding the recommended guidance of MARSSIM for Class 1 and 2 structures, we request that you provide to the NRC in advance of performing the survey, the technical basis used to justify the larger survey unit size. The NRC will conduct a review of your technical basis to determine if an adequate justification has been documented supporting the increase in survey unit size and will determine if an NRC confirmatory survey of the area will be necessary. This issue is discussed further in Section 1.2 of this inspection report.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Dwight D. Chamberlain, Director
Division of Nuclear Materials Safety

Docket No.: 50-344
License No.: NPF-1

Enclosure: NRC Inspection Report
50-344/01-02

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-344
License No.:	NPF-1
Report No.:	50-344/01-02
Licensee:	Portland General Electric Company
Facility:	Trojan Nuclear Plant
Location:	71760 Columbia River Highway Rainier, Oregon 97048
Dates:	May 14 -17 and August 7 and 22, 2001
Inspectors:	E. M. Garcia, Health Physicist R. S. Clement, Sc.D., Health Physicist, NMSS T. J. Vitkus, CHP, ORISE S. X. Schneider, Senior Health Physicist, NMSS
Approved by:	D. Blair Spitzberg, Ph.D., Chief Fuel Cycle & Decommissioning Branch
Attachment:	1) Supplemental Information 2) List of Documents Reviewed 3) ORISE Report, "Confirmatory Survey of the Containment Dome, Trojan Nuclear Plant, Rainier, Oregon," dated August 2001
ADAMS Entry:	IR 05000344-01-02; on 05/14/2001-05/17/2001; Portland General Electric Co.; Trojan Nuclear Plant; Decommissioning Report; No violations.

EXECUTIVE SUMMARY

Trojan Nuclear Plant
NRC Inspection Report 50-344/01-02

The licensee had developed, approved and issued procedures and training outlines which described the process to perform final status surveys at the Trojan site in accordance with the license termination plan approved by the NRC on February 12, 2001, and issued as License Amendment 206. The licensee had completed the final radiological survey of the internal surface of the reactor building containment dome on May 8, 2001. Results of the final survey were approved by the licensee as a final status survey data package on May 16, 2001.

The NRC's contractor, Oak Ridge Institute for Science and Education (ORISE), provided support to review the licensee's implementation of the final survey program and to perform confirmatory survey activities on the interior surface of the containment dome. The results of this independent survey confirmed the results of the licensee's survey and supported the conclusion that the reactor building containment dome's interior surface was below approved derived concentration guideline levels (DCGLs).

Inspection of Final Surveys

- Procedures to implement the final survey program had been approved and issued. Training for the final survey program had been completed (Section 1).

- Confirmatory surveys for the containment dome, including surface scans, surface activity measurements, and smear sampling were performed by the Oak Ridge Institute for Science and Education (ORISE) for the NRC. All results were below the DCGL and confirmed the adequacy of the licensee's final survey results. An Inspection Follow-up Item was identified to review the technical basis used to justify future survey unit sizes which exceed the recommended survey unit sizes contained in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Section 1).

Follow-up of Open Items

- The licensee's actions relating to 16 inspection follow-up items identified in Inspection Report 50-344/00-03 were reviewed. Fifteen items were closed (Section 2).

Report Details

Summary of Facility Status

The Trojan nuclear power station was operated until November 1992 and entered permanent shutdown status January 1993. The major dismantlement activities and remediation effort at the Trojan site have been completed. On February 12, 2001, the NRC issued Amendment 206 to the Trojan license approving the license termination plan. This plan described the process to be implemented over the next several years by Portland General Electric to conduct a comprehensive site survey of the facilities, structures and site areas to confirm that radiation levels remaining after site remediation are below regulatory limits specified in 10 CFR Part 20, Subpart E. Confirmation that radiological levels are below Subpart E limits at the facility is necessary to support eventual termination of the Part 50 reactor license.

The spent reactor fuel currently remains in wet storage in the spent fuel pool. The licensee plans to establish an Independent Spent Fuel Storage Installation (ISFSI) at the Trojan site for dry cask storage of the spent fuel under a 10 CFR Part 72 license. Movement of the spent fuel to the ISFSI will be completed prior to the Part 50 license being terminated.

1 Inspection of Final Surveys (83801)

1.1 Inspection Scope

The final survey program for the Trojan site was currently underway. The procedures and training program to support this effort were reviewed. Confirmatory measurements were performed by the NRC contractor of the interior surface of the containment dome to confirm the licensee's final survey results.

1.2 Observation and Findings

The licensee had developed numerous procedures to implement the final survey program. A list of procedures reviewed during this inspection is provided as Attachment 2 to this inspection report. All procedures were final and had been approved. The procedures were consistent with the commitments in the license termination plan.

The training modules developed for the final survey program provided an adequate level of detail and content. A training matrix had been developed identifying four training groups and six training modules. Review of attendance records confirmed completion of the training of personnel on April 30, 2001. The final status surveys began on May 1, 2001.

The site data management system (SDMS) was tested to verify the statistical comparisons performed by the software that would result in a survey unit failing or the need to perform additional investigation. The SDMS appropriately flagged each specified statistical quantity that failed the comparison.

Radiological survey personnel from the Oak Ridge Institute for Science and Education (ORISE), under contract to the NRC, provided support to review the licensee's implementation of the final survey program and to perform confirmatory surveys of the interior surface of the containment dome. Document reviews, interviews with licensee staff and surveyors, surface scans, surface activity measurements, and smear sampling was conducted. The ORISE report, "Confirmatory Survey of the Containment Dome, Trojan Nuclear Plant, Rainier, Oregon," dated August 2001, is provided as Attachment 3 to this inspection report. All results of surveys conducted of the containment dome were below the derived concentration guidelines levels (DCGL) and confirmed the adequacy of the licensee's final survey results.

Surface scans of the containment dome indicated relatively uniform residual beta activity with no "hot spots" of contamination. Surface activity measurements for total beta activity ranged from 3,300 dpm/100 cm^2 to 12,000 dpm/100 cm^2. Removable activity was measured up to 2 dpm/100 cm^2 alpha radiation and 15 dpm/100 cm^2 beta radiation.

Statistical analysis of the final survey data showed that the survey unit standard deviation (σ) was within 20 percent of the mean. The mean and median of the data were essentially the same which further supports uniformity of residual activity.

Eight locations were selected for performing comparative side-by-side measurements between the licensee and the NRC. Results of the side-by-side measurements agreed to within 5 percent. The licensee's data acquisition and field survey techniques were observed on a selected area of the containment building wall with activity ranging from approximately 2,000 cpm to 5,000 cpm over the 126 cm^2 area of the detector. Scan speeds, detector-to-surface distance, and area coverage were appropriately demonstrated by the survey personnel.

The containment dome had been classified as a Class 2 structure by the licensee. This classification was determined to be appropriate by the NRC. However, the size of the survey area assigned to the containment dome of 1,910 m^2 exceeded the suggested maximum Class 2 survey unit area size of 1,000 m^2 recommended in Section 4.6 of the MARSSIM. A technical basis for increasing the survey unit size had been developed by the licensee. Although MARSSIM and the approved license termination plan allow survey units to be sized to preserve dose modeling, neither document provided specific guidance for increasing survey unit sizes from those recommended in MARSSIM. In the case of the containment dome, the following concerns were identified by the NRC inspectors related to the size of the survey unit and the technical basis developed by the licensee:

- For the dose modeling used by the licensee, a non-uniform contamination model for the walls and ceiling of 50 percent and 10 percent, respectively, as compared to the floor contamination, was assumed. Relative dose rates using a non-uniform wall, ceiling and floor distribution model of 50 percent, 10 percent, and 100 percent ratios is comparable to the uniform distribution "infinite plane" model shown in NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning," Vol. 1, Figure 6.1. The "infinite plane" model is used in MARSSIM for the building occupancy scenario. The "infinite plane" model resembles the non-uniform distribution model but does not closely resemble the uniform distribution model. The licensee's assumption that the contamination was distributed in the containment building in a non-uniform distribution was not supported by any actual measurements. The assumptions used for the uniformity of the contamination distribution in the room can significantly affect the dose calculations for the building occupancy scenario.

- MARSSIM recommends survey unit sizes that include a reasonable sampling density. If the survey unit size is too large, an inadequate number of samples may be taken to statistically support the conclusion that an area surveyed is below the DCGL.

The NRC has determined that for the containment dome, no additional surveys will be required despite the size of the survey unit exceeding the MARSSIM recommendation. This determination was based on the results of the NRC's confirmatory survey conducted during this inspection. Both the NRC survey and the licensee's survey confirmed that residual contamination was well below the DCGLs and scans did not identify any areas approaching the DCGLs. Also, the licensee's operational survey program had established an investigation level of 5,000 dpm/100 cm^2 for contamination surveys. This operational survey level was very conservative and provided reasonable assurance that the potential for residual contamination being present, yet not detected, would be very low.

However, future confirmatory surveys by the NRC will only include representative areas of the licensee's facility and will not include all areas of the Trojan facility. Therefore, for future surveys at Trojan using survey unit sizes exceeding the recommended guidance in Section 4.6 of MARSSIM for Class 1 and 2 structures, the NRC is requesting that the licensee provide to the NRC in advance of performing the survey, the technical basis used to justify the larger survey unit size. The NRC will conduct a review of the technical basis to determine if an adequate justification has been documented supporting the increase in survey unit size. The NRC will also determine if an NRC confirmatory survey of the area will be necessary. This item will be tracked as Inspection Follow-up Item (50-344/0102-01).

1.3 Conclusion

Procedures to implement the final survey program had been approved and issued. Training for the final survey program had been completed.

Confirmatory surveys for the containment dome, including surface scans, surface activity measurements, and smear sampling were performed by ORISE for the NRC. All results were below the DCGL and confirmed the adequacy of the licensee's final survey results. An Inspection Follow-up Item was identified to review the technical basis used to justify future survey unit sizes which exceed the recommended survey unit sizes contained in MARSSIM.

2 Follow-up of Open Items (92701)

2.1 <u>(Closed) IFI 50-344/0003-04: Instrument Set-up, Performance Check, and Investigation Level</u>. Procedure RP 456, "Control and Operation of Data Logger Survey Instruments," Rev. 1, had been approved and issued. This procedure described the process for setting up an instrument to collect final status survey data. The procedure required response checks prior to issue and after the download of survey data to the SDMS. The procedure also required instruments to be response checked in a reproducible geometry or on a flat surface as appropriate. If the instrument's response check was within 20 percent of the mean background and net count rate, the instrument was considered acceptable for survey data collection.

2.2 <u>(Closed) IFI 50-344/0003-05: Calibration Procedures and NIST Traceability</u>. The licensee had approved and issued procedure RP 459, "Final Survey Instruments Calibration Program," Rev. 0, and RP 456, "Control and Operation of Data Logger Survey Instruments," Rev. 1. These procedures included provisions for source energies, surface efficiencies and environmental effects during instrument calibrations. The procedures also required the performance of instrument response checks at the beginning and end of each data acquisition period. The licensee was using National Institute of Standards and Technology (NIST) traceable sources for the calibrations and performance checks.

2.3 <u>(Closed) IFI 50-344/0003-06: Bar Codes</u>. The licensee had approved and issued lessons plan FS-S-33-LP, "Survey Instrumentation & Data Collection," Rev. 1, which addressed the potential concern for selecting an incorrect bar code identification number (ID) when a detector had more than one ID number. The final version of the lesson plan was reviewed and found to include specific instructions concerning the bar code process for selection of the correct detector/instrument/cable length combination in Section C, "Using Data Logger Survey Instruments." Several of the licensee's surveyors and the State of Oregon inspector, who participated in the training, were interviewed to verify an understanding of the bar code selection process and were found to be knowledgeable of the correct method for selecting the ID number.

2.4 <u>(Discussed) IFI 50-344/0003-07: Scan and Static MDCs and Instrument and Surface Efficiencies</u>. The licensee had determined an instrument efficiency (ε_i) of 0.41 using a NIST traceable technetium-99 wide-area flood source (source number T-788) to calibrate the Ludlum Model M2350-1 data logger (Serial Number 142499) with the Model 43-68 surface contamination detector (Serial Number 148630) used in the final status survey of the containment dome. A jig was used by the licensee to calibrate the detector in a reproducible geometry with a 1 cm source-to-detector distance. The ε_i value considered a correction factor of 1.26 to account for the detector's physical area.

The surface efficiency (ε_s) value of 0.5 for beta measurements on smooth surfaces was approved in the licensee's license termination plan. The inspectors performed an independent assessment of the weighted beta energy for the radionuclide mixture in RPC 2001-01, which provided the technical basis for implementing the gross activity derived concentration guideline level ($DCGL_{GA}$) of 22,000 dpm/100 cm^2 gamma/beta for the final status survey of the containment dome. A weighted maximum beta energy greater than 400 keV was calculated by the inspectors for the radionuclide mixture. This supported the licensee's ε_s value of 0.5 for this maximum beta energy. A technical basis for determining a site-specific ε_s value for measuring contamination of irregular or uneven surfaces (e.g., scabbled concrete, embedded piping) had not been developed because these types of measurements were not required for the containment dome. Appropriate ε_s value(s) for irregular or uneven surfaces will need to be developed for final status surveys in the fuel and auxiliary buildings, if these types of surfaces are present.

The inspectors performed an independent calculation of the instrument total efficiency (ε_T). The calculated ε_T values ranged from 0.19 counts per disintegration (cpd) to 0.21 cpd, which compared favorably with the licensee's ε_T value of 0.21 for the same make and model detector.

One of the primary procedures used by the licensee to conduct surveys was RP 452, "Final Survey Data Collection," Rev. 4. This procedure included specific instructions on survey techniques to be used by the personnel performing the surveys. This procedure was reviewed and discussed with the licensee. Side-by-side surveys were conducted by the licensee and the ORISE staff on May 16, 2001. As a result of the review of this procedure and observation of the side-by-side surveys, it was determined that inadequate guidance was provided concerning when an investigation should be conducted if the scan MDC was exceeded during a survey. The licensee had calculated a scan MDC of 740 dpm/100 cm^2. This was much lower than the investigation level of 22,000 dpm/100 cm^2 (1.0 × DCGL) and below the alarm set point or action level of 16,500 dpm/100 cm^2 (0.75 × DCGL). According to Section 5.2 of RP 452, an investigation survey was required when the investigation level of 22,000 dpm/100 cm^2 was exceeded. However, when a survey indicated a reading between the scan MDC and the investigation level, guidance was not provided in the procedure or training program on what actions should be taken.

Procedure RP 452 also instructed the survey personnel to use the audio divide scale on the data logger instrument to enhance the ability to detect an increasing count rate. However, because the background count rate is also reduced when the audio divide scale is used, a new scan MDC would need to be calculated. Further concerns were also noted related to survey personnel not using the instrument's audio output during scanning near the boundary of a survey area to detect possible increases in activity levels. In these situations, adjacent areas to the survey area may not be selected for further scanning because the increased activity was not recognized.

The licensee stated their intent to review procedure RP 452 to determine if a revision of the procedure and additional training of survey personnel was needed to address 1) actions to take when the scan MDC is exceeded, 2) how to use the audio divide scale during surveys, and 3) use of audio output during scanning near the boundary of a survey area. Procedure and training changes made by the licensee related to these issues will be reviewed during a future inspection.

2.5 <u>(Closed) IFI 50-344/0003-08: Survey Unit Surface Areas for Building Surfaces</u>. The licensee had approved and issued procedure RP 451, "Final Survey Unit Design," Rev. 2, to describe the methodology for designing final status survey units. This procedure followed MARSSIM guidance for classifying areas of contamination potential, maintaining conventional size restrictions, and specifying scanning coverage for impacted survey units. The procedure also described site-specific scan and static investigation levels and the process for reclassifying survey units, if needed, during the final status survey.

Section 4.3.4.a. of RP 451 stated that, "A survey unit that has been remediated is classified as Class 1 or Class 2." The NRC finds this acceptable when remediation was conducted for As Low As Reasonably Achievable (ALARA) purposes only and the residual radioactivity concentrations in Class 2 survey units does not exceed the derived concentration guideline level (DCGL) prior to remediation.

2.6 <u>(Closed) IFI 50-344/0003-09: Background Subtractions for Surface Activity Measurements</u>. Not subtracting background is the correct method for measuring surface activity when performing the wilcoxon rank sum test. The licensee had approved and issued procedure RP 452, "Final Survey Data Collection," Rev. 4. This procedure was reviewed to ensure it did not direct the user to perform background subtraction from the gross surface activity measurements. The procedure referenced procedure RP 456, "Control and Operation of Data Logger Survey Instruments," Rev. 1, and procedure RP 459, "Final Survey Instruments Calibration Program," Rev. 0. The final approved versions of these two procedures were also reviewed and were found to not direct the user to subtract background.

2.7 <u>(Closed) IFI 50-344/0003-12: Radionuclide Variability and Data Quality Assessment</u>. The final approved versions of procedure RP 453, "Final Survey Data Processing," Rev. 1 and procedure RP 460, "Final Survey Data Management," Rev. 0, were reviewed. These procedures adequately addressed data variability and the data quality assessment process. The actual standard deviation (σ) value for the containment surveys was found to be 564.9 dpm/100 cm^2, which equated to 0.03 x DCGL. Since this was less than the procedure RP 453 default value of σ greater than or equal to 0.2 $\times$ DCGL, then no statistical hypothetical testing of the containment data and no additional measurements were required.

2.8 <u>(Closed) IFI 50-344/0003-13: Area Factors for Building Surface Contamination</u>. The licensee had approved and issued procedure RP 457, "Development and Application of Derived Concentration Guideline Levels," Rev. 0, to incorporate the NRC approved area factors for evaluating elevated residual radioactivity in structural surfaces and soils in Class 1 survey units.

2.9 <u>(Closed) IFI 50-344/0003-16: Cross-Check Program and Evaluation of Results</u>. Inspection Report 50-344/00-03 documented that the licensee had participated in a cross-check program with an independent laboratory. For one sample analyzed, the lab results between the licensee and the offsite laboratory could not be compared since the licensee's results were expressed on a sample basis (μCi/sample) and not in meaningful radiological units (e.g., μCi/g). The licensee has since conducted a review of the situation surrounding this sample. The sample in question was a scraping from a pipe having a mass of 0.206 grams. The licensee did not have an appropriate calibration geometry for this small sample and thus, the results were reported as activity per sample. In 1998, when the sample was collected and analyzed, the licensee had compared six other sample results with the offsite lab. In 1999, three samples were compared and in 2000, one sample was compared. For all samples, the licensee's analysis results were within the acceptance criteria of plus or minus 20 percent.

2.10 (Closed) IFI 50-344/0003-17: Training Program. All required initial training was completed on April 30, 2001, prior to beginning final status surveys on May 1, 2001.

2.11 (Closed) IFI 50-344/0003-18: User Manual, Procedures and DCGLs for Site Data Management System. The licensee had approved and issued procedure RP 457, "Development and Application of Derived Concentration Guideline Levels," Rev. 0, procedure RP 460, "Final Survey Data Management," Rev. 0, and procedure RG 20-10, "SDMS Application Instructions," Rev. 0. The procedures were reviewed and found to be adequate. The approved $DCGL_{GA}$ value of 22,000 dpm/100 cm^2 gamma/beta for the final status survey of the containment dome had been input into the SDMS.

2.12 (Closed) IFI 50-344/0003-19: Independent Testing of Site Data Management System. A data set was input into the SDMS computer using static surface activity measurements from a wall in the containment building selected by the NRC to independently test the SDMS. The data set was edited to observe how the SDMS performed its statistical comparisons of the range, median, mean, and standard deviation (σ) as approved in the license termination plan and described in the licensee's approved and issued procedure RP 453, "Final Survey Data Processing," Rev. 1. The sample data, instrument service history, data uploading report, data testing, screening exception report, exception report, and statistical quantities reports generated from the independent testing were reviewed. The data set was correctly analyzed by the SDMS.

2.13 (Closed) IFI 50-344/0003-20: Internal QA Audit and Corrective Actions. The licensee had issued an internal memorandum entitled, "Followup QA Surveillance of Part 50 Final Survey Plan Readiness," 01-005-SURV, dated April 30, 2001. This memorandum documented the licensee's Quality Assurance (QA) surveillance of the Part 50 Final Survey Plan Readiness performed on March 6 - April 26, 2001. Two of the 13 open items in the report were reviewed and discussed with the licensee.

The first open item reviewed involved embedded piping. The licensee had planned to perform evaluations of the embedded piping whenever:

- the average (mean) measured residual radioactivity from building surface contamination containing embedded piping resulted in a total effective dose equivalent (TEDE) of greater than or equal to 20 mrem/yr ($0.8 \times$ DCGL), and/or

- the mean measured residual radioactivity from inside the embedded piping results in a TEDE greater than or equal to 5 mrem/yr (100,000 dpm/100 cm^2 beta/gamma)

These limits were established to ensure that the TEDE for the non-embedded and embedded portions of a survey unit, when combined, would not exceed 25 mrem/yr. In a teleconference with NRC staff on April 10 and 12, 2001, the licensee provided several examples (contrived test cases) for a Class 1 survey unit to clarify the methodology for data point investigations and sample population evaluations for embedded piping surveys based on the mean DCGL in a survey unit. The methodology was determined to be acceptable.

The second item involved the plotting of data. Application of posting and frequency plots were described in Section 4.6.2, "Graphical Data Review," of the licensee's approved license termination plan, and in Section 2.4, "Final Site Survey" of the NRC staff's safety evaluation report (SER) dated February 12, 2001. The licensee had approved and issued procedure RP 453, "Final Survey Data Processing," Rev. 1. In Section 4.4 of RP 453, the licensee had planned to generate a frequency plot (histogram) for all survey data sets. A posting plot would also be generated, but only as part of an investigation survey whenever one or more static measurements exceed the DCGL for a survey data set. Although MARSSIM, Section 8.2.2.2 recommends both posting and frequency (histogram) plots of all survey data sets, the licensee's proposal to generate plots as described in RP 453 was determined to be adequate because the licensee's operational survey and ALARA program goal limits of 5,000 dpm/100 cm^2 beta/gamma for surface contamination were conservative. The authors of the NRC Safety Evaluation Report concurred with this conclusion.

2.14 <u>(Closed) IFI 50-344/0003-22: Linking Instructions to Data Base Files</u>. The licensee had approved and issued procedure RP 460, "Final Survey Data Management," Rev. 0, for maintaining SDMS applications and the database, and radiation protection guideline RG 20-10, "SDMS Applications Instructions," Rev. 0, for storage, processing and reporting of final survey data. The final versions of RP 460 and RG 20-10 provided adequate directions concerning documenting additional survey instructions, when provided to survey teams, into the SDMS data base as a means of recording any special directions given to the team related to a particular survey unit.

2.15 <u>(Closed) IFI 50-344/0003-23: Inclusion of Background Values on Data Sheets</u>. The licensee had approved and issued procedures RP 452, "Final Survey Data Collection," Rev. 4, and RP 456, "Control and Operation of Data Logger Survey Instruments," Rev 1. The final versions of RP 452 and RP 456 provided correct instructions related to the subtraction of background and had deleted the directions provided in the draft version of the procedures to subtract a local background from the gross readings measured. The approved procedure is consistent with MARSSIM for applying the wilcoxon rank sum test.

2.16 <u>(Closed) IFI 50-344/0003-24: EPA/DOE Interlaboratory Comparison Test Failures</u>. The licensee had participated in the Environmental Protection Agency (EPA) and Department of Energy (DOE) inter-laboratory comparison program for the environmental sampling program conducted at the Trojan site. This involved sending split samples to the EPA/DOE labs and the offsite vendor laboratory used by Trojan for counting environmental samples. Soil, water and filter samples were sent to the labs in October and December 1999. Of the 50 samples, one-third failed the inter-comparison tests due to excessive differences between the results of the licensee's vendor laboratory and the EPA/DOE laboratory.

The licensee conducted an evaluation of the high number of failures. The licensee concluded that the reason for the failures was the change in key personnel at the offsite vendor laboratory. The offsite vendor has taken corrective actions to resolve the problem. Selected samples collected in 1999 were recounted and did not require the licensee to modify and resubmit their 1999 Environmental Report. The year 2000 inter-comparison tests did not have a high number of failures. The licensee will continue to monitor results provided by the vendor laboratory.

3 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management and staff at the exit meeting on May 17, 2001. Telephonic conversations were conducted on August 7 and August 22, 2001, to further discuss the issue concerning the survey unit size for the containment dome survey. The licensee did not indicate that any of the information presented at the exit meeting was proprietary.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

L. Dusek, Nuclear Regulatory Affairs Manager
T. Meek, Radiation Protection Manager
G. Huey, Radiation Protection Supervisor
J. Cooper, Radiation Protection Engineer
L. Rocha, Health Physicist
M. Stein, Health Physicist

State of Oregon

A. Bless, Oregon Office of Energy

INSPECTION PROCEDURES USED

83801	Inspection of Final Surveys
92701	Follow-up of Open Items

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-344/0102-01	IFI	Review the technical basis used to justify future survey unit sizes which exceed the recommended survey unit sizes contained in MARSSIM.

Discussed

50-344/0000-07	IFI	Scan and Static MDCs and Instrument and Surface Efficiencies

Closed

50-344/0003-04	IFI	Instrument Set-up, Performance Check, and Investigation Level
50-344/0003-05	IFI	Calibration Procedures and NIST Traceability
50-344/0003-06	IFI	Bar Codes
50-344/0000-08	IFI	Survey Unit Surface Areas for Building Surfaces
50-344/0003-09	IFI	Background Subtractions for Surface Activity Measurements
50-344/0003-12	IFI	Radionuclide Variability and Data Quality Assessment
50-344/0003-13	IFI	Area Factors for Building Surface Contamination
50-344/0003-16	IFI	Cross-Check Program and Evaluation of Results
50-344/0003-17	IFI	Training Program
50-344/0003-18	IFI	User Manual, Procedures and DCGLs for Site Data Management System
50-344/0003-19	IFI	Independent Testing of Site Data Management System
50-344/0003-20	IFI	Internal QA Audit and Corrective Actions
50-344/0003-22	IFI	Linking Instructions to Data Base Files
50-344/0003-23	IFI	Inclusion of Background Values on Data Sheets

50-344/0003-24	IFI	EPA/DOE Interlaboratory Comparison Test Failures

LIST OF ACRONYMS

CFR	Code of Federal Regulations
DandD	Decontamination and Decommissioning Computer Code
DCGL	Derived Concentration Guideline Level
$DCGL_{GA}$	Gross Activity DCGL
IFI	Inspector Follow-up Item
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
QA	Quality Assurance
RP	Radiation Protection Procedure
SDMS	Site Data Management System
TPP	Trojan Plant Procedure
WRS	Wilcoxon Rank Sum (statistical test)

ATTACHMENT 2

LIST OF DOCUMENTS REVIEWED

Procedures Reviewed

- TPP 14-24, "Operational Survey Unit Guidelines and Transfer of Turnover Units," Rev. 0.
- RP 450, "Final Survey Administrative and Quality Controls," Rev. 2.
- RP 451, "Final Survey Unit Design," Rev. 2.
- RP 452, "Final Survey Data Collection," Rev. 4 (QR).
- RP 453, "Final Survey Data Processing," Rev. 2.
- RP 454, "Final Survey Background Reference Areas," Rev. 2 .
- RP 455, "Final Survey Quality Control Measurements," Rev. 2.
- RP 456, "Control and Operation of Data Logger Survey Instruments," Rev. 1 (QR).
- RP 457, "Development and Application of Derived Concentration Guideline Levels," Rev. 0.
- RP 458, "Final Survey Remediation Levels and ALARA Evaluations," Rev. 0.
- RP 459, "Final Survey Instruments Calibration Program," Rev. 0 (QR).
- RP 460, "Final Survey Data Management," Rev. 0 (QR).

Lesson Plans Reviewed

- FS-S-11-LP "Final Survey Plan Overview Training (w/exam)".
- FS-S-22-LP "Final Survey Unit Design".
- FS-S-33-LP "Survey Instrumentation and Data Collection (w/exam)".
- FS-S-55-LP "Scan and Static Survey Practical".
- FS-S-66-LP "Final Survey Data Processing".
- FS-S-77-LP "Survey Data Management System (SDMS) Overview (w/quiz)".

Technical Basis Documents Reviewed

- RPC 2001-01 "Gross Activity DCGL for Containment" (QR).
- RPC 2001-04 "Final Survey Technical Basis Document: Sizing of Survey Units Consisting of Non-Floor Surface Areas" (QR).

ATTACHMENT 3

CONFIRMATORY SURVEY OF THE CONTAINMENT DOME
TROJAN NUCLEAR PLANT

AUGUST 2001

CONFIRMATORY SURVEY
OF THE
CONTAINMENT DOME
TROJAN NUCLEAR PLANT
RAINER, OREGON

T.J. Vitkus and J.R. Morton

Prepared for the
U.S. Nuclear Regulatory Commission
Headquarters Office

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

CONFIRMATORY SURVEY
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CONTAINMENT DOME
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Prepared by

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FINAL REPORT

AUGUST 2001

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

CONFIRMATORY SURVEY
OF THE
CONTAINMENT DOME
TROJAN NUCLEAR PLANT
RAINIER, OREGON

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ACKNOWLEDGMENTS

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ABBREVIATIONS AND ACRONYMS

ϵ_i	instrument efficiency
ϵ_s	source efficiency
ASME	American Society of Mechanical Engineers
cm	centimeter
cm^2	square centimeter
cpm	counts per minute
DCGL	derived concentration guideline level
DOE	U.S. Department of Energy
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
EML	Environmental Measurements Laboratory
ESSAP	Environmental Survey and Site Assessment Program
ISFSI	Independent Spent Fuel Storage Installation
ITP	Intercomparison Testing Program
m^2	square meter
MAPEP	Mixed Analyte Performance Evaluation Program
MDC	minimum detectable concentration
MeV	million electron volts
MWe	megawatt electric
MWt	megawatt thermal
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NRIP	NIST Radiochemistry Intercomparison Program
ORISE	Oak Ridge Institute for Science and Education
PGE	Portland General Electric
TNP	Trojan Nuclear Plant

**CONFIRMATORY SURVEY
OF THE
CONTAINMENT DOME
TROJAN NUCLEAR PLANT
RAINIER, OREGON**

INTRODUCTION AND SITE HISTORY

The Trojan Nuclear Plant (TNP), located in Rainier, Oregon, was jointly owned by the Portland General Electric (PGE), the City of Eugene, the Eugene Water and Electric Board, and Pacific Light/PacifiCorp and achieved initial criticality in December 1975. It began commercial operation in May 1976. The nuclear steam supply system was a four-loop pressurized water reactor that created an output rated at 3,411 MWt, with an approximate net electrical output rating of 1,130 MWe. After 17 years of operation, 14 fuel cycles, and approximately 3,300 effective full power days, the TNP was shut down for the final time on November 9, 1992 because of a steam generator tube leak precipitated by a failed sleeve. On January 27, 1993, PGE notified the U. S. Nuclear Regulatory Commission (NRC) of its decision to permanently cease power operations and decommission the facility (PGE 1999).

The decommissioning of the TNP was divided into two broad periods: a Transition Period and a Decontamination and Dismantlement Period. The Transition Period began with the plant shutdown in January of 1993 and will continue until the removal of the spent fuel from its temporary location in the spent fuel pool and movement to the Independent Spent Fuel Storage Installation (ISFSI). The Decontamination and Dismantlement Period and Transition Period are being performed concurrently. Contaminated systems, components, and structural materials have been or are currently being decontaminated or removed. Final status surveys have been initiated on several surfaces and structures. The reactor vessel, together with internal components, has been removed and shipped off-site for disposal. Final status surveys have recently been completed on the reactor containment dome. The primary radionuclides of concern are mixed activation and fission products with Cs-137 as the predominant radionuclide.

Due to PGE's desire to remove the polar crane and the fact that the crane provides the best access to the containment dome, the NRC's Headquarters Office requested that the Oak Ridge Institute for Science and Education's (ORISE), Environmental Survey and Site Assessment Program (ESSAP) perform a confirmatory survey on the TNP reactor containment dome and to provide technical assistance with an in-process inspection.

SITE DESCRIPTION

The TNP facility is located on 634 acres along the Columbia River in Columbia County, Oregon, approximately 42 miles north of Portland near the city of Rainier (Figure 1). The Reactor Containment Building, together with the Turbine, Auxiliary, Fuel and other support buildings are located within the site's radioactive control area (Figure 2). The containment dome itself consisted of the area encompassed between the 197'-7" and 249'-9" elevations with an approximate area of 1,900 m^2, and a surface composed of painted steel (Figure 3).

OBJECTIVES

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

DOCUMENT REVIEW

ESSAP reviewed the licensee's instrument calibration and derived concentration guideline level calculations and final radiological survey data for adequacy and appropriateness.

PROCEDURES

ESSAP performed a confirmatory survey of the containment dome during the period of May 15 and 16, 2001. Survey activities were conducted in accordance with a site-specific survey plan and the ORISE/ESSAP Survey Procedures and Quality Assurance Manuals (ORISE 2001a, 2000a and b). Additional information concerning major instrumentation, sampling equipment, and survey and analytical procedures may be found in Appendices A and B.

REFERENCE SYSTEM

PGE classified the containment dome as a single, Class 2 survey unit then sub-divided the containment dome into 18 segments. These segments were based on the ring girders that are spaced at 20-degree intervals around the circumference of the polar crane track. ESSAP randomly selected six of the segments—1, 3, 5, 10, 14 and 15—for survey and used this system to reference survey locations.

SURFACE SCANS

Approximately 50 percent of the selected containment dome section surfaces were scanned for gamma and beta activity using NaI scintillation and gas proportional detectors. Particular attention was given to attachments, weldings, and other locations where material may have accumulated. All detectors were coupled to ratemeters or ratemeter-scalers with audible indicators.

SURFACE ACTIVITY MEASUREMENTS

Direct measurements for total beta activity were performed at 10 locations on the containment dome surface using gas proportional detectors coupled to portable ratemeter-scalers (Figure 4 through 8). Of the ten measurement locations, five were at PGE measurement points. Smear samples, for determining removable activity levels, were also collected from each direct measurement location.

Normally, ESSAP procedure requires collecting background direct measurements on a suitable reference material. The background count rates are then used in the survey unit measurement conversion to net surface activity levels. PGE has elected to conservatively represent final status survey results without a background correction. Therefore, with NRC concurrence, the decision was made to adapt the PGE approach for confirmatory measurements to permit a more direct comparison of the ESSAP and PGE results.

IN-PROCESS INSPECTION

ESSAP reviewed the licensee's survey area classification and documentation, data quality assessment process, statistical analyses, findings in support of its final status surveys, and observed survey procedures.

ESSAP and PGE surface activity determinations were compared by performing eight direct one-minute measurements of varying levels of radioactivity within the containment building.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ORISE's ESSAP laboratory in Oak Ridge, Tennessee for analysis and interpretation. Samples were analyzed in accordance with the ESSAP Laboratory Procedures Manual (ORISE 2001b). Smears were analyzed for gross alpha and gross beta activity using a low background gas proportional counter. Smear data and direct measurements for surface activity were converted to units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2).

FINDING AND RESULTS

SURFACE SCANS

Surface scans of the containment dome showed the presence of relatively-uniform elevated residual beta activity. Surface scans did not identify any "hot spot" contamination above the relatively uniform elevated activity.

SURFACE ACTIVITY LEVELS

Results of total and removable activity for the containment building dome are summarized in Table 1. The total beta activity levels ranged from 3,300 to 12,000 dpm/100 cm^2. Removable activity ranged from 0 to 2 dpm/100 cm^2 for alpha and -2 to 15 dpm/100 cm^2 for beta. ESSAP measurements compared favorably with those reported by PGE at similar locations (Table 2).

IN-PROCESS INSPECTION ITEMS

As part of the confirmatory process, several items of PGE's final status survey program were evaluated. Specific areas addressed included 1) survey area classification and corresponding survey coverage, 2) instrument calibration, 3) data quality assessment and statistical analyses, and 4) observation of field survey techniques and comparative measurements. The evaluation results were as follows:

1. The review of PGE characterization and final status survey data, coupled with confirmatory survey results showed that PGE had appropriately classified the containment dome as Class 2. The area of the survey unit, 1,900 m^2, exceeded the suggested maximum Class 2 survey unit area of 1,000 m^2. PGE scanned approximately 10% of the total survey unit area. This coverage satisfies the minimum MARSSIM-recommended scanning coverage and appears appropriate based on the uniform distribution of residual activity. Because the area should have been designated as two separate survey units due to the size, PGE surface scan data were evaluated to ensure that all areas received adequate scanning coverage. Each of the 18 containment dome segments had a total area of 12 m^2 scanned out of the available 110 m^2. PGE performed 35 direct measurements using a random-start systematic pattern. This number of measurements exceeded the number necessary for the statistically-based survey design requirements (based on the estimated contaminant variability and Type I and Type II decision errors of 0.05). However, PGE may not have met the survey commitments made in their license termination plan—taking into account the fact that the containment dome

should have consisted of two survey units—where a minimum of 30 measurements were required per survey unit.

2. The PGE detector calibration consisted of a weighted beta energy determination based on the radionuclide mixture present and the corresponding beta energy distribution. ESSAP performed an independent validation of the beta energy calculation with acceptable results. The calculated ESSAP total efficiencies ranged from 0.19 to 0.21 and PGE reported a total efficiency of 0.21 calibrated to Tc-99. The total efficiency included a surface efficiency factor of 0.5 that is most representative of the beta energy emissions of the predominant radionuclide, Cs-137. Additional information is provided in Appendix B. Both groups used Ludlum Model 43-68 gas proportional detectors.

3. The review of the PGE data quality assessment process validated the PGE database and assessment process. Basic statistical analysis of the containment dome final status survey data showed retrospectively that the survey unit standard deviation was within the site-specific goal of less than 20% of the mean. The mean and median of the data were essentially the same which further supports the absence of significant hot spots. Scan results were compared with required action levels, with no activity identified in excess of the action level. Furthermore, the data management software was challenged with direct measurement data collected from an area of the containment building that had not yet been remediated. The reason for selecting a contaminated area was to ensure that residual surface activity variability was greater than the 20% PGE-estimated variance, that elevated measurements would be present, and the mean activity would exceed PGE- required values. The initial review of the challenge data determined that there were multiple criteria that would result in either the failure of the survey unit or indicate that additional investigations would be required. The data management system appropriately flagged each criterion. One discrepancy with the downloaded data was noted where a direct measurement was not recorded.

4. ESSAP observed PGE's data acquisition and field survey techniques and performed side-by-side direct measurements. ESSAP selected eight locations for performing comparative direct measurements. The results of the measurement comparison are provided in Table 2. Results agreed within ± 5%.

   ESSAP next selected an area of the containment building wall with activity ranging from approximately 2,000 to 5,000 counts per minute over the 126 cm^2 physical detector area for observation of PGE scanning techniques. ESSAP observed PGE's technicians and determined that scan speeds, detector to surface distance, and area coverage were appropriate. It was not apparent that the survey procedures require listening to the audio output to identify elevated activity. Rather, PGE had established an alarm set point for their data logger instrument that corresponded to the PGE-required investigation level. The radioactivity was distributed in the selected scanning area in such a manner that multiple alarms should have occurred. However, the instrument only alarmed once when initially crossing the region where the activity in excess of the action level was located. No further alarm events were observed. PGE reported that the reason no additional alarms occurred was because the instrument may not have been appropriately set up. Discussions determined that the instrument only records the highest reading identified during scans of a specified area. Based on these observations, ESSAP questions the applicability of the MDC_{scan} provided in the PGE Survey Instrument MDC Data Sheet. This form reported a calculated MDC_{scan} of 740 dpm/100 cm^2 developed from MARSSIM guidance for human factors performance. However, if an alarm event is the only means employed to cause either the surveyor to stop and investigate an area or for an area to be investigated after reviewing the downloaded data, this MDC_{scan} may not be appropriate. Furthermore, a related procedural question also was identified—whether or not survey personnel will investigate elevated activity that may exist outside of a specified scanning area.

Additional review is therefore recommended to ensure that scan areas with multiple locations of elevated activity are adequately investigated as it appeared to the observer that PGE relies on an instrument alarm and retrospective data analysis, rather than the audio output of the instrument, to identify areas of elevated activity. Also, the implementation of the procedure for bounding areas of elevated activity should be evaluated.

COMPARISON OF RESULTS WITH GUIDELINES

The contaminants of concern for this site are beta-gamma emitters resulting from the operation of the TNP facility, with Cs-137 as the primary radionuclide for the containment dome. PGE's NRC-approved gross activity DCGL for interior surfaces of the reactor containment is 22,000 dpm/100 cm^2. No direct measurements exceeded this guideline. A site-specific removable activity guideline has not been developed. When screening level DCGLs are adapted, the assumption is that the removable fraction is equal to 0.1 (2,200 dpm/100 cm^2) (FR 1998). Removable activities were all less than or equal to the MDC of the procedure and therefore satisfied this condition.

SUMMARY

At the request of the Nuclear Regulatory Commission's Headquarters Office, the Environmental Survey and Site Assessment Program of the Oak Ridge Institute for Science and Education conducted a confirmatory survey of the containment building dome at the Trojan Nuclear Plant in Rainier, Oregon. Confirmatory activities performed on May 15 and 16, 2001 included reviews of PGE final status survey process and data, confirmatory surface scans, direct measurements, and sampling for removable contamination.

The results of the confirmatory activities verified the radiological conditions of the containment dome reported by PGE. Surface activity levels were below the DCGL. In-process inspection items were overall satisfactory, but with additional reviews recommended of field survey and investigation procedures.

FIGURES

788-001 (1)

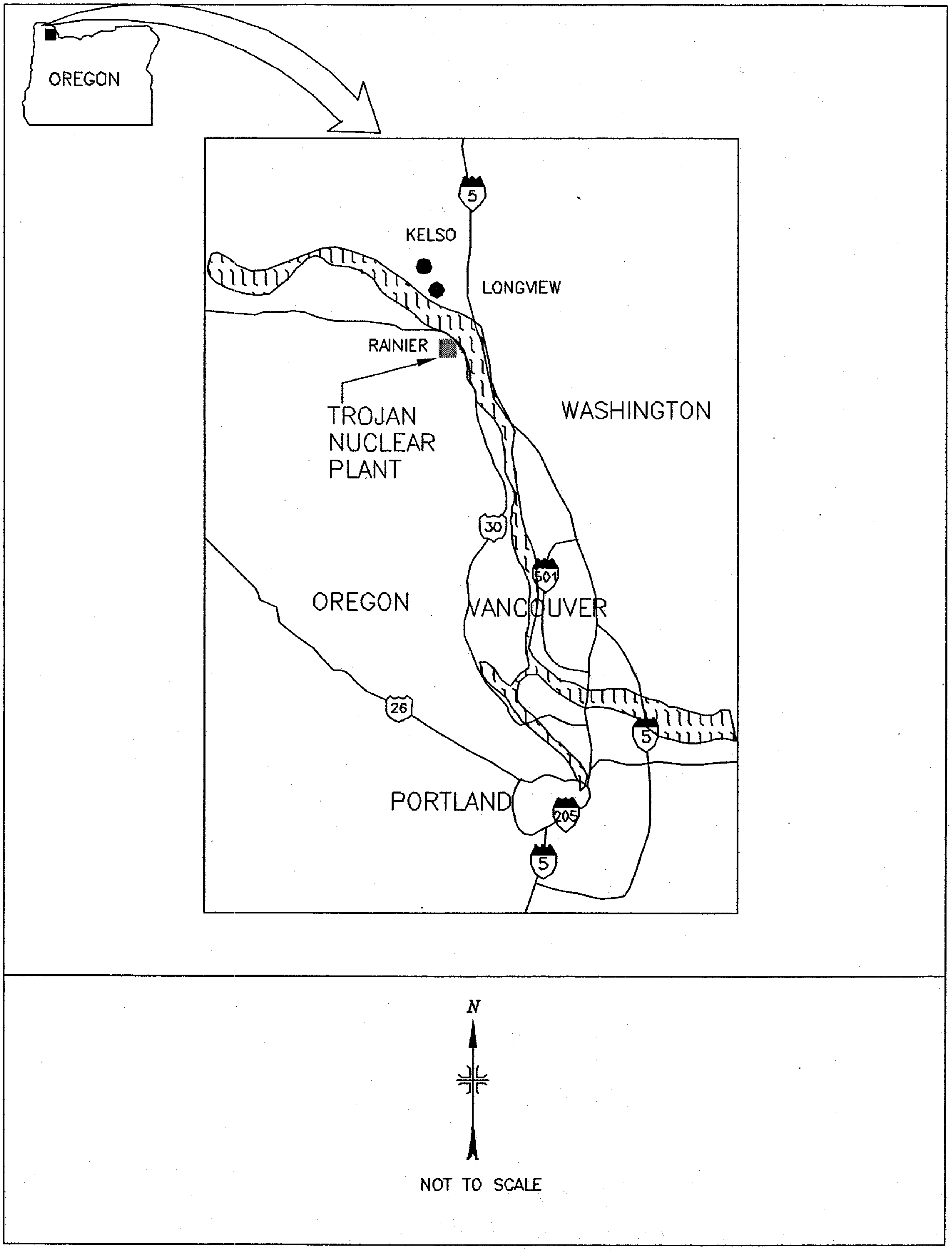

FIGURE 1: Location of the Trojan Nuclear Plant

788–002 (1)

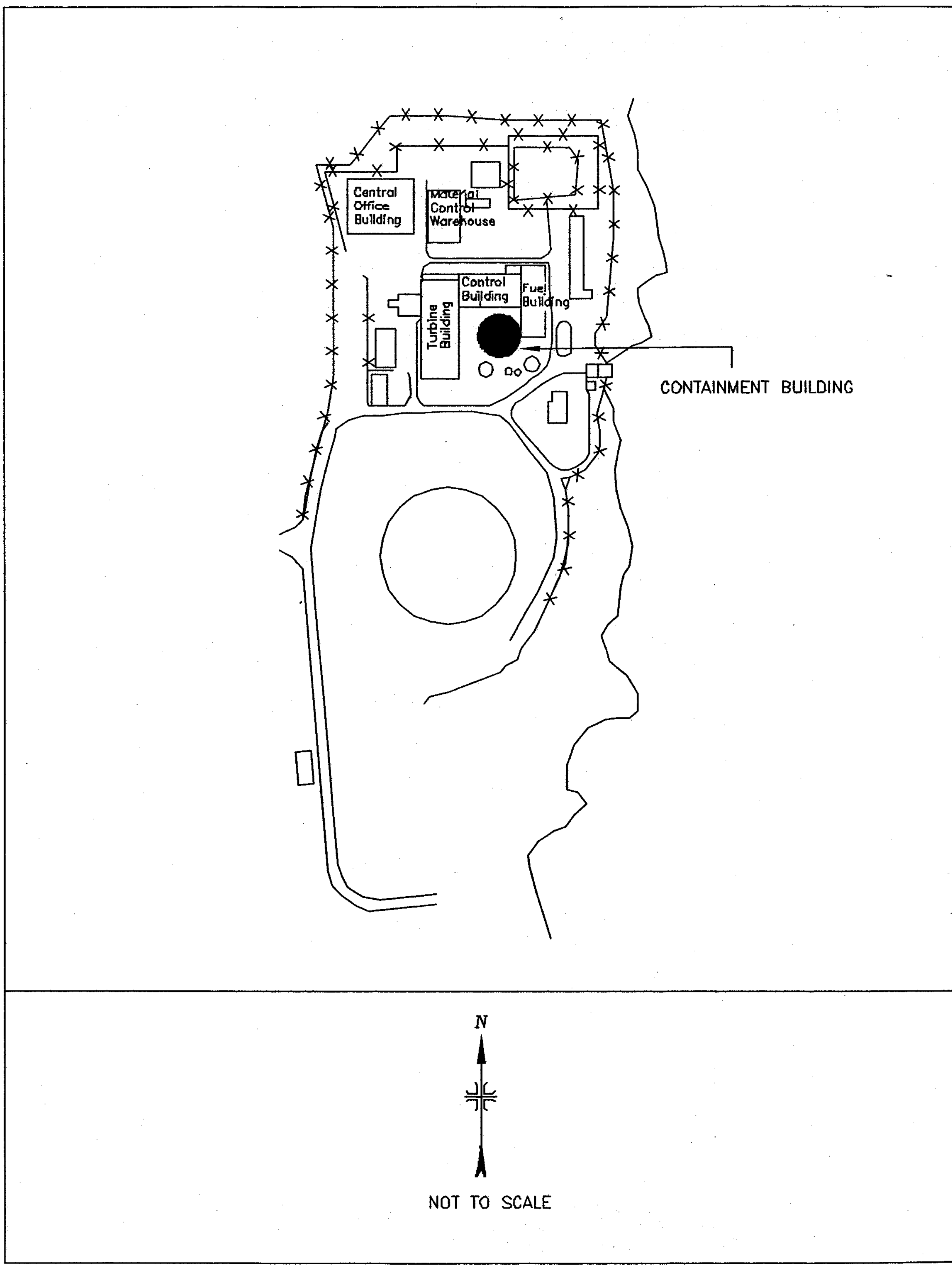

FIGURE 2: Trojan Nuclear Plant – Location of Containment Building

788-008 (1)

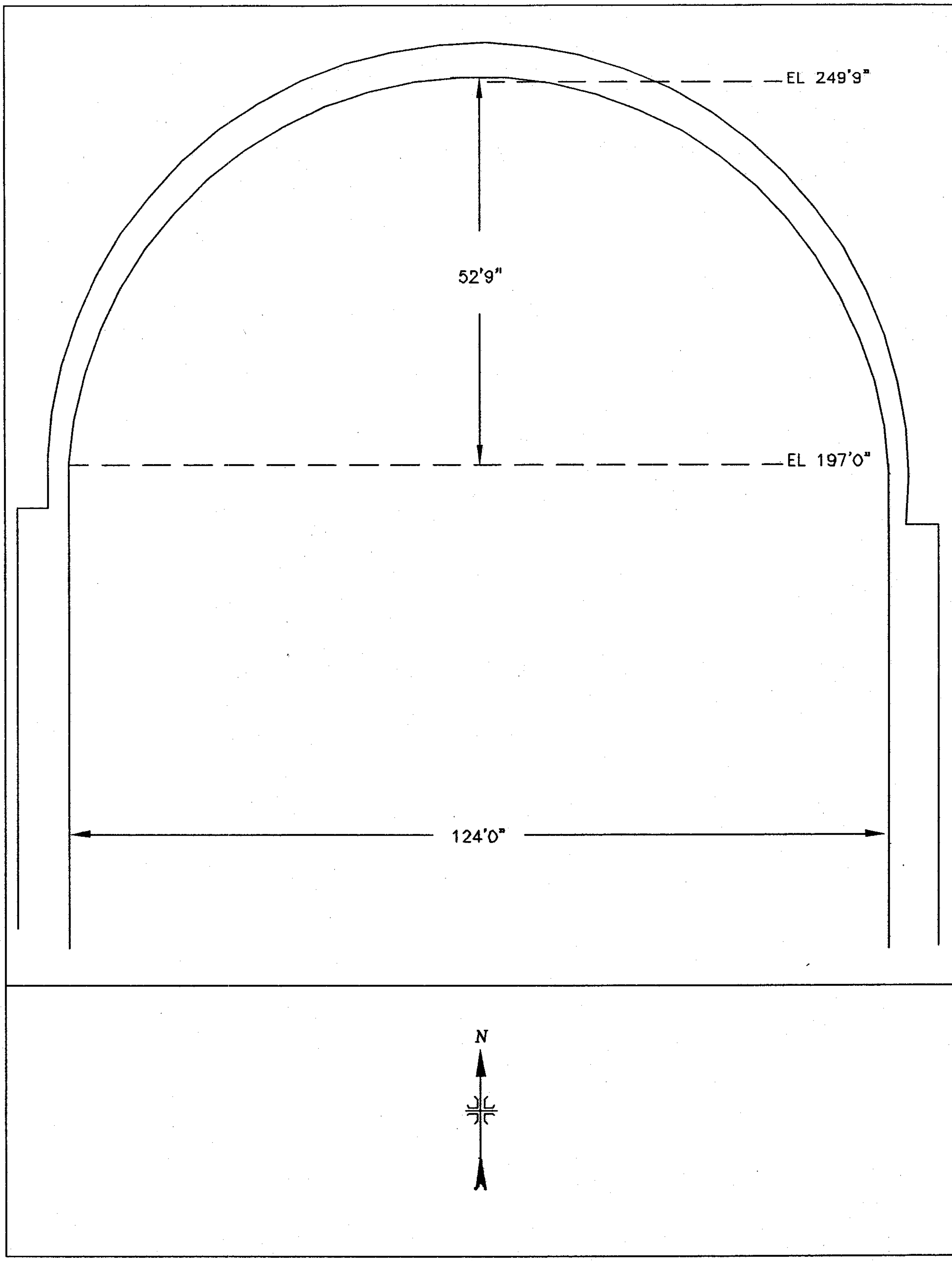

FIGURE 3: Plot Plan Trojan Containment Dome

788-003(2)

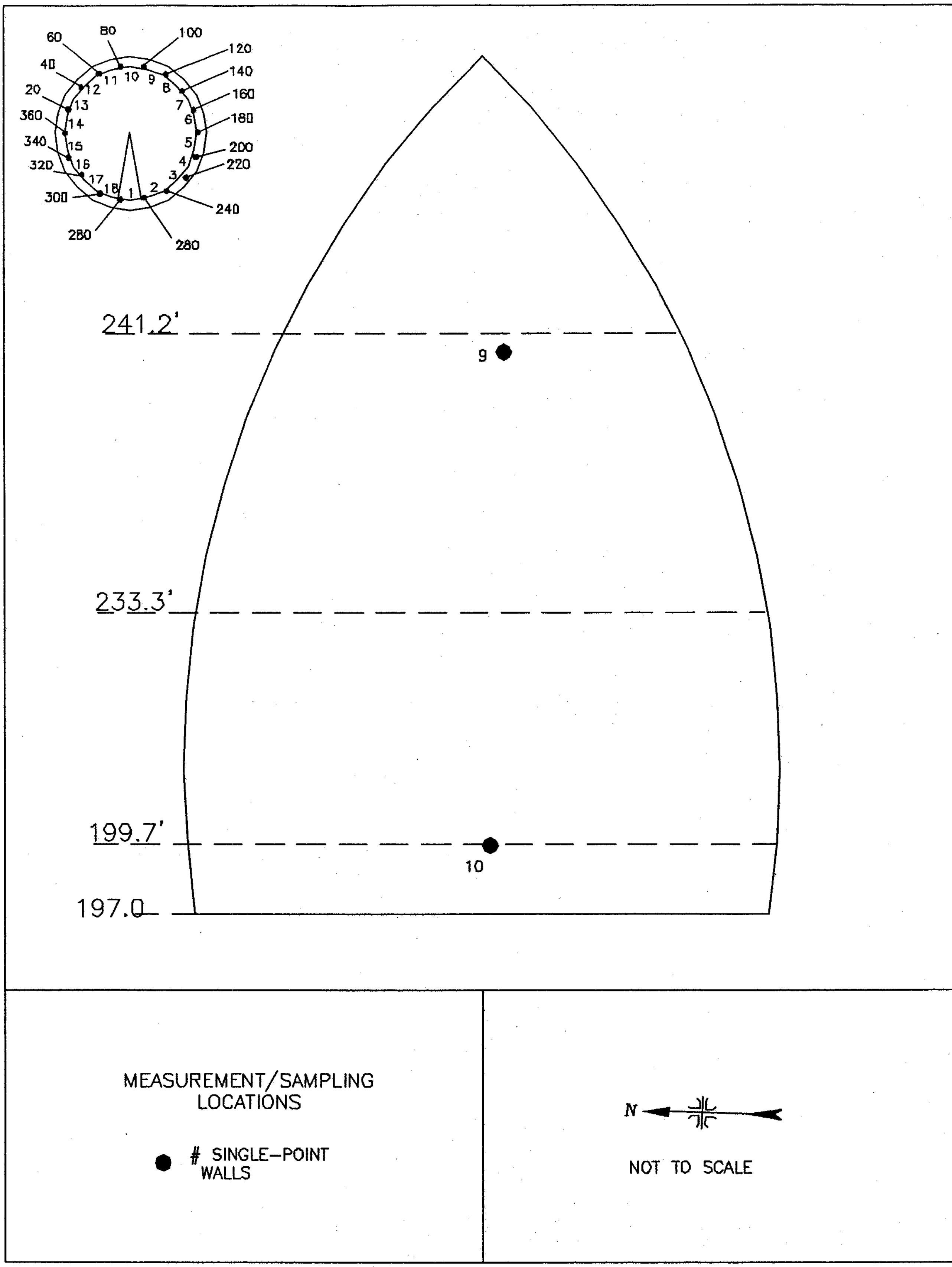

FIGURE 4: Containment Building Dome, Section I – Measurement and Sampling Locations

788–004 (1)

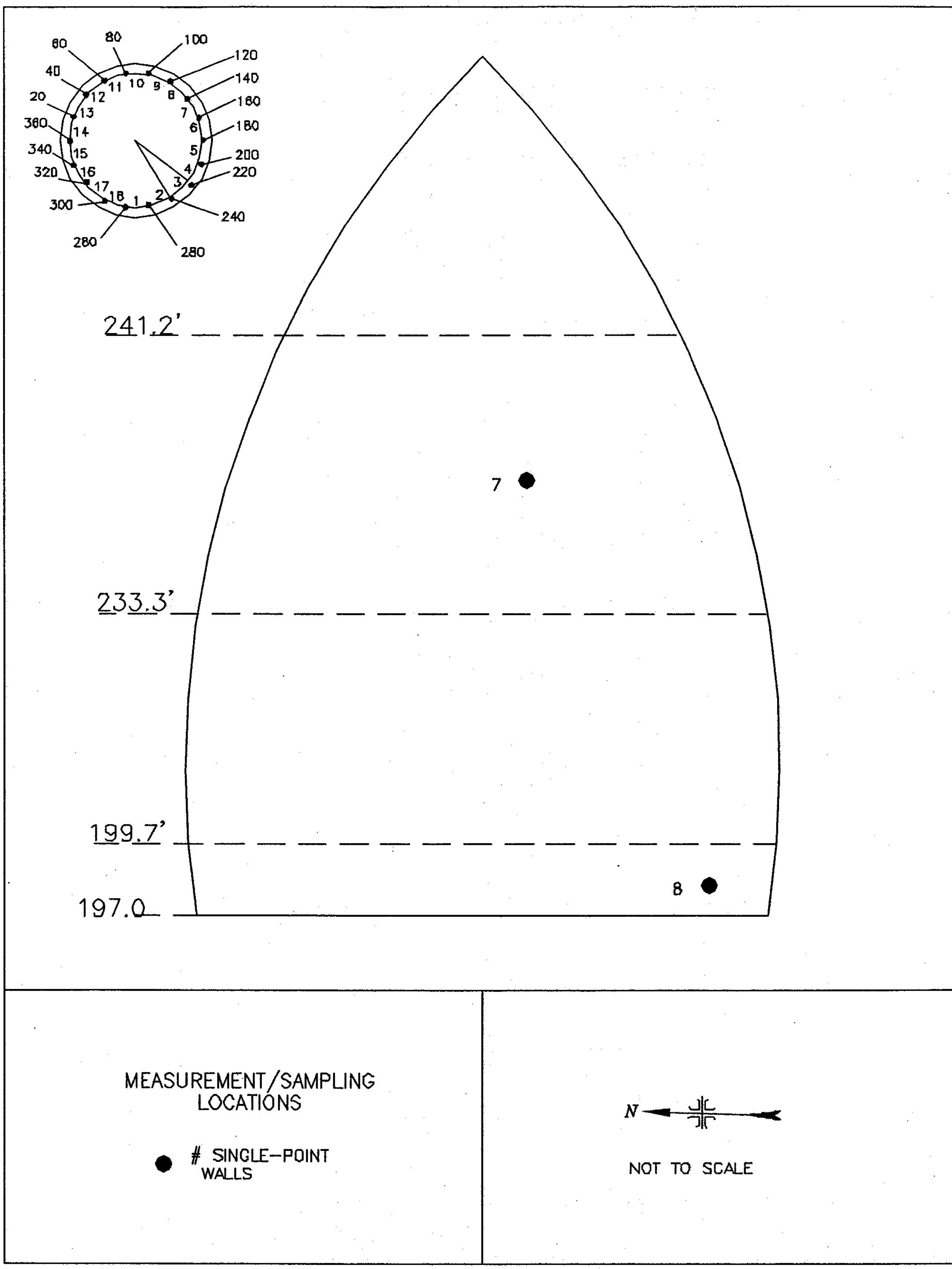

FIGURE 5: Containment Building Dome, Section 3 – Measurement and Sampling Locations

788–005 (1)

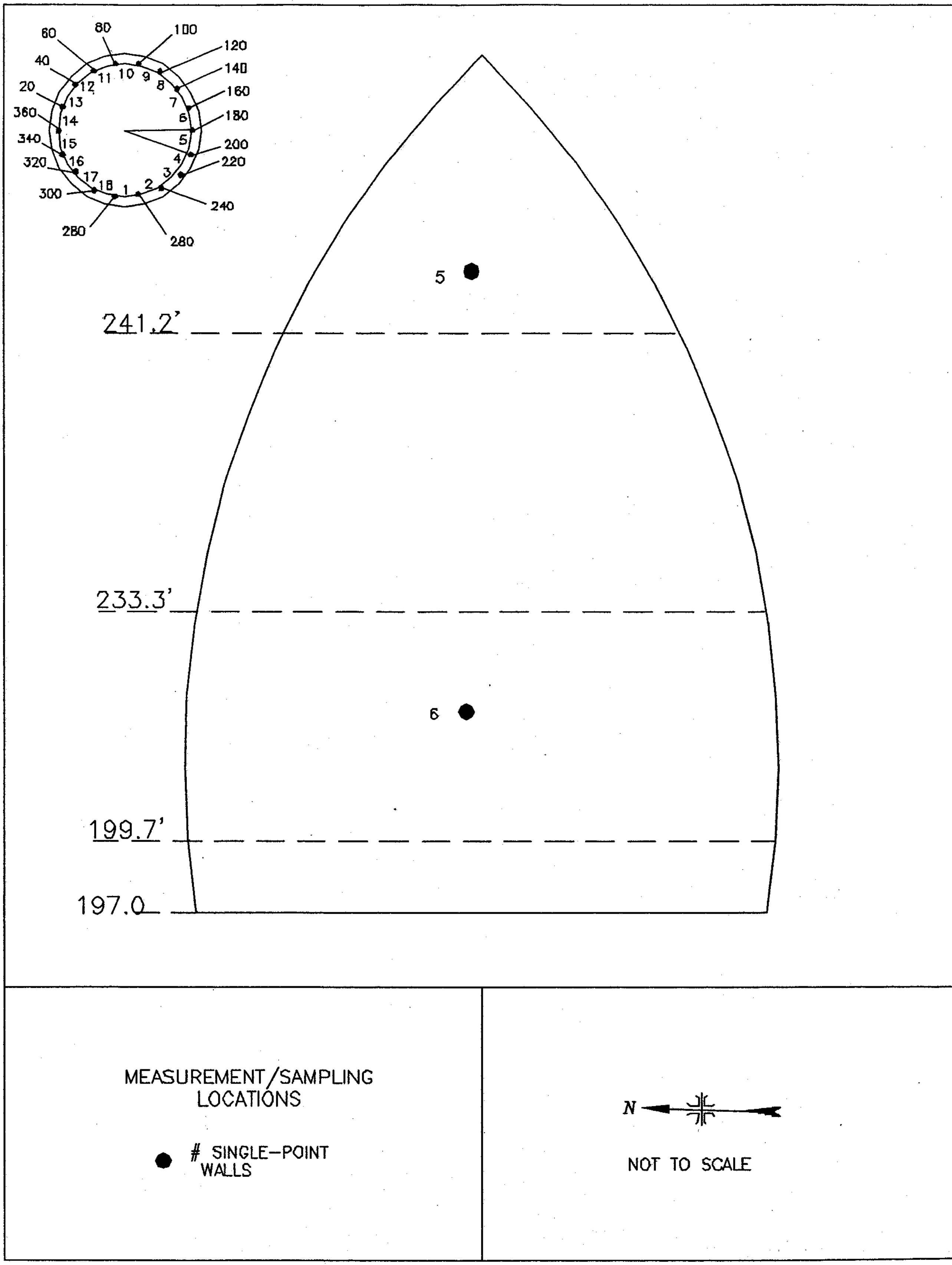

FIGURE 6: Containment Building Dome, Section 5 – Measurement and Sampling Locations

788-006 (1)

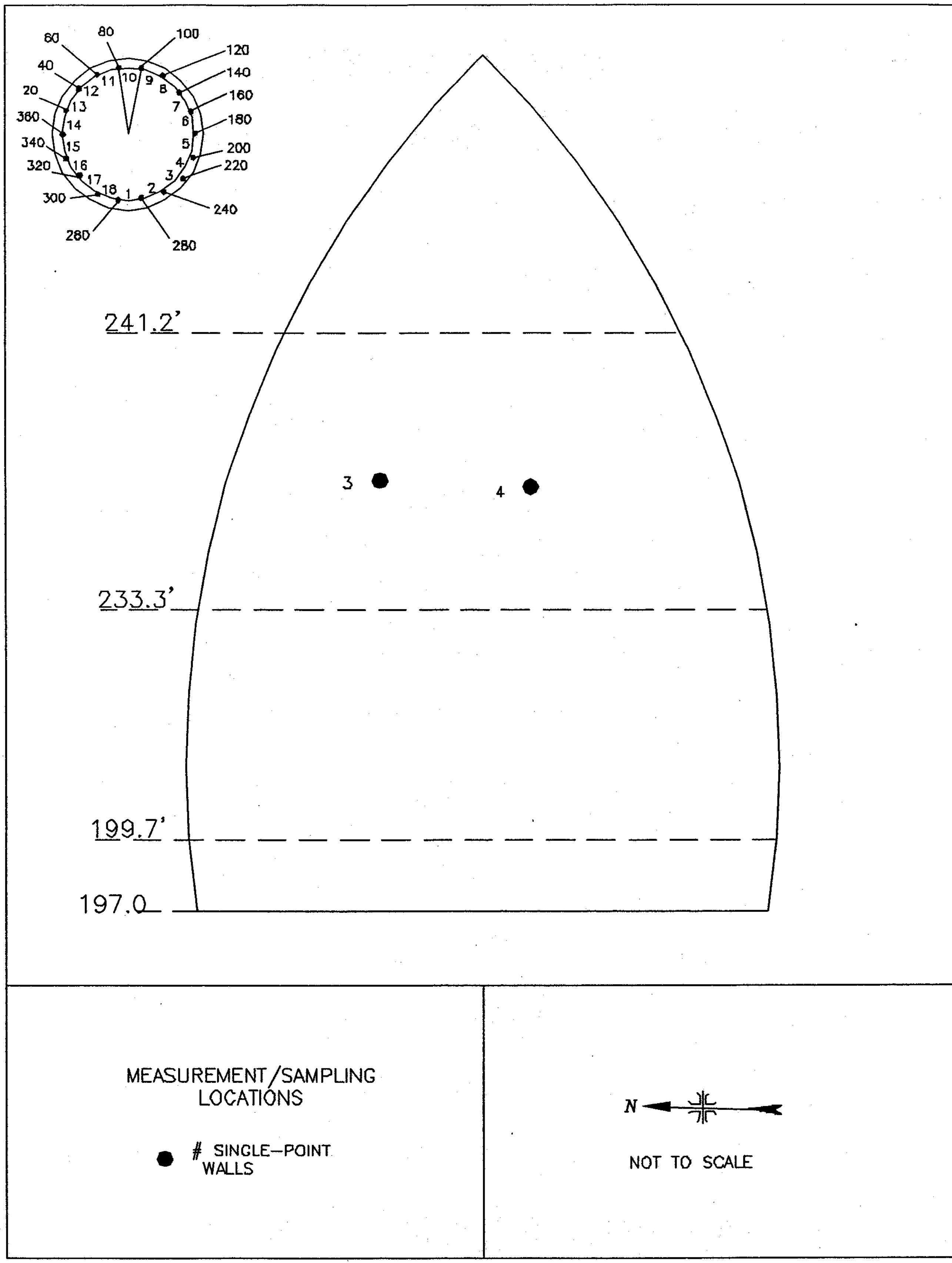

FIGURE 7: Containment Building Dome, Section 10 – Measurement and Sampling Locations

788-007 (x)

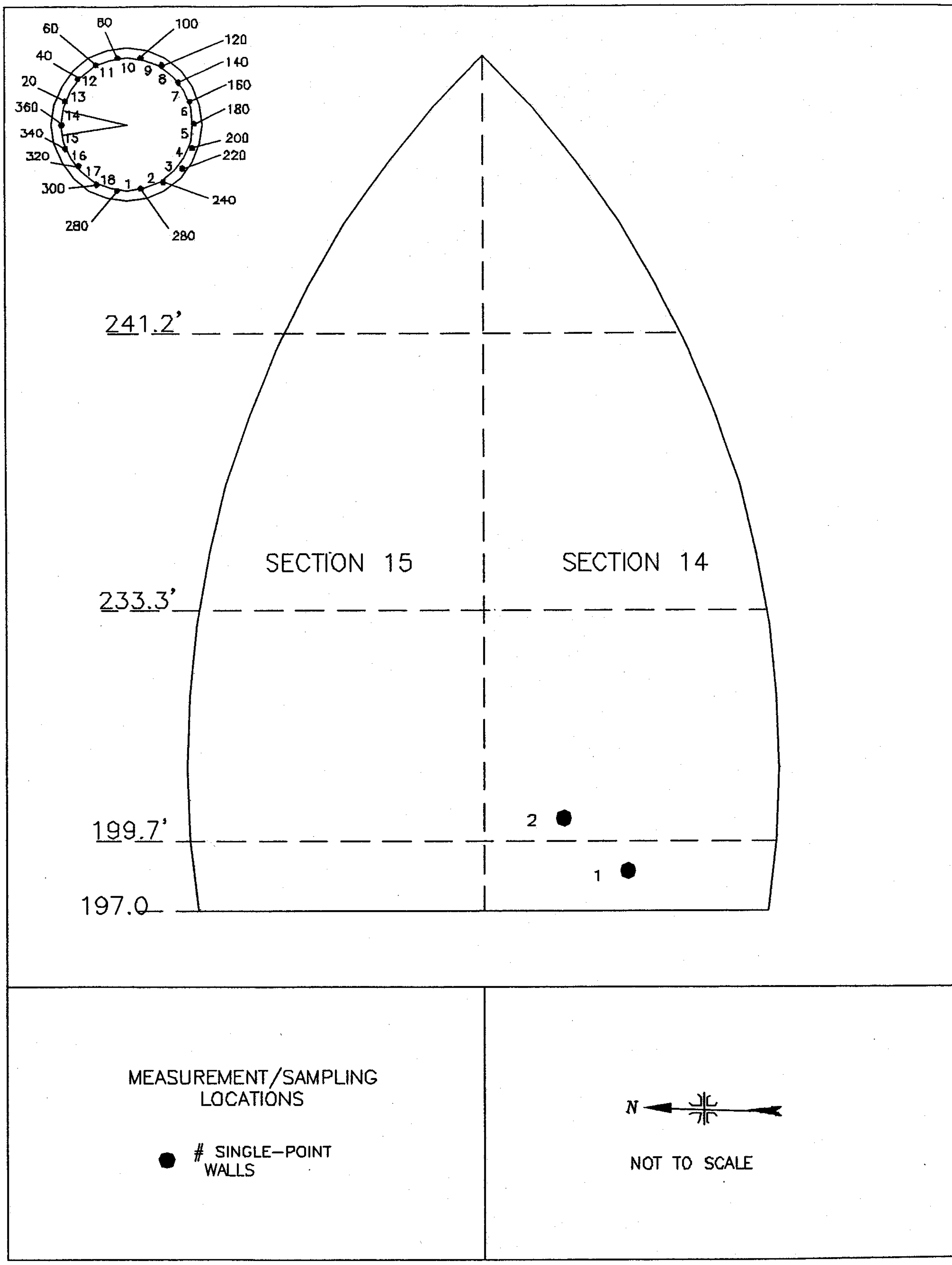

FIGURE 8: Containment Building Dome, Sections 14 and 15 –
Measurement and Sampling Locations

TABLES

TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS
CONTAINMENT DOME
TROJAN NUCLEAR PLANT
RAINER, OREGON

Location[a]	Total Beta Activity (dpm/100 cm^2)[b]		Removable Activity (dpm/100 cm^2)	
	ORISE	PGE	Alpha	Beta
1[c]	3,300	3,500	0	3
2	4,000	NA	0	3
3[d]	5,600	4,800	0	1
4	12,000	NA	0	9
5	11,000	NA	0	15
6	3,700	NA	0	6
7[e]	3,300	2,900	2	1
8[f]	2,900	3,000	0	-2
9	4,200	NA	0	4
10[g]	3,900	4,000	0	3

[a]Refer to Figures 4 through 8.
[b]Surface activity levels are gross values. No background subtraction performed.
[c]TNP Location #24.
[d]TNP Location #13.
[e]TNP Location #33.
[f]TNP Location #18.
[g]TNP Location #35.

TABLE 2

SURFACE ACTIVITY LEVELS—COMPARATIVE MEASUREMENTS
CONTAINMENT BUILDING
TROJAN NUCLEAR PLANT
RAINIER, OREGON

Location	Total Beta Activity (dpm/100 cm^2)[a]	
	ORISE	PGE
Containment Building, 45' Level		
1	87,000	87,000
2	8,900	8,600
3	55,000	54,000
4	24,000	24,000
5	36,000	37,000
6	19,000	20,000
7	10,000	9,600
8	29,000	28,000

[a]Surface activity levels are gross values. No background subtraction performed.

REFERENCES

Federal Register (FR): Volume 63, No. 222, Page 64134; November 18, 1998.

Oak Ridge Institute for Science and Education (ORISE). Survey Procedures Manual for the Environmental Survey and Site Assessment Program. Oak Ridge, TN; 2000a.

Oak Ridge Institute for Science and Education. Quality Assurance Manual for the Environmental Survey and Site Assessment Program. Oak Ridge, TN; 2000b.

Oak Ridge Institute for Science and Education. Confirmatory Survey Plan for the Containment Dome, Trojan Nuclear Plant, Rainier, Oregon (Docket No. 50-344, RFTA No. 01-008). Oak Ridge, TN; May 11, 2001a.

Oak Ridge Institute for Science and Education. Laboratory Procedures Manual for the Environmental Survey and Site Assessment Program. Oak Ridge, TN; 2001b.

Pacific General Electric (PGE). Trojan Nuclear Plant License Termination Plan. Rainier, Oregon; August 1999.

APPENDIX A

MAJOR INSTRUMENTATION

APPENDIX A
MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the authors or the authors' employer.

DIRECT RADIATION MEASUREMENT

Instruments

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

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

Detectors

Ludlum Gas Proportional Detector
Model 43-68
Physical Probe Area, 126 cm^2
(Ludlum Instruments, Inc., Sweetwater, TX)

Victoreen NaI Scintillation Detector
Model 489-55
3.2 cm x 3.8 cm crystal
(Victoreen, Cleveland, Ohio)

LABORATORY ANALYTICAL INSTRUMENTATION

Low Background Gas Proportional Counter
Model LB-5100-W
(Canberra/Tennelec, Oak Ridge, TN))

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

APPENDIX B
SURVEY AND ANALYTICAL PROCEDURES

PROJECT HEALTH AND SAFETY

All survey and laboratory activities were conducted in accordance with ORISE health and safety and radiation protection programs.

CALIBRATION AND QUALITY ASSURANCE

Calibration of laboratory instrumentation was based on standards/sources, traceable to NIST, when such standards/sources were available. In cases where they were not available, standards of an industry-recognized organization were used.

Analytical and field survey activities were conducted in accordance with procedures from the following documents of the Environmental Survey and Site Assessment Program:

- Survey Procedures Manual (September 2000)
- Laboratory Procedures Manual (May 2001)
- Quality Assurance Manual (March 2000)

The procedures contained in these manuals were developed to meet the requirements of Department of Energy (DOE) Order 414.1A 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, ITP, and EML Laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

SURVEY PROCEDURES

Surface Scans

Surface scans were performed by passing the detectors slowly over the surface; the distance between the detectors and the surface was maintained at a minimum—nominally less than 1 cm. Identification of elevated levels was based on increases in the audible signal from the recording and/or indicating instrument. Combinations of detectors and instruments used for the scans were:

Gamma	-	NaI scintillation detector with ratemeter
Alpha - Beta	-	gas proportional detector with ratemeter-scaler

Scan minimum detectable concentrations (MDCs) were estimated using the calculational approach described in NUREG-1507.[1] The MDC_{scan} is a function of many variables, including the background level. Beta background levels ranged from 260 to 280 cpm for the hand-held gas proportional detector. Additional parameters selected for the calculation of MDC_{scan} 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. The instrument efficiencies (ϵ_i) for the hand-held gas proportional detectors calibrated to Tc-99 ranged from 0.38 to 0.41. To illustrate an example for the hand-held gas proportional, the minimum detectable count rate (MDCR) and scan MDC can be calculated as follows:

$$s_i = d'\sqrt{b_i} = 4.92\ counts$$

$$MDCR = (s_i)\left(\frac{60}{i}\right) = 295\ cpm$$

where s_i = minimum number of net source counts in the interval
b_i = number of background counts in the observation interval (i) = (270 counts minute^{-1}) (i seconds/60 seconds minute^{-1})

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

The MDC_{scan} is calculated assuming a source efficiency (ϵ_s) of 0.5 and instrument efficiency (ϵ_i) of 0.38:

$$MDC_{scan} = \frac{MDCR}{\sqrt{0.5}(\varepsilon_s)(\varepsilon_i)} = 2{,}200\ dpm / 100\ cm^2$$

Surface Activity Measurements

Measurements of total surface activity levels were performed using gas proportional detectors with portable ratemeter-scalers. Surface activity measurements were made on structural steel. 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 gross rate by the total efficiency ($\epsilon_i \times \epsilon_s$) and correcting for the active area of the detector. No background correction was made in order to accurately compare ESSAP results with PGE's. The 2π instrument efficiency factors (ϵ_i) ranged from 0.38 to 0.41 for the gas proportional detectors calibrated to Tc-99. The source efficiency factor (ϵ_s) was 0.5. The total beta efficiency factors for gas proportional detectors were 0.19 and 0.21. Tc-99 was selected as the calibration source as it provides a conservative representation of the beta energy distribution of the radionuclide mixture, consisting of predominantly Cs-137, reported for the site. ISO-7503[2] recommends an ϵ_s of 0.25 for beta emitters with a maximum energy of less than 0.4 MeV and an ϵ_s of 0.5 for maximum beta energies greater than 0.4 MeV. Although the beta energy of the calibration source is less than the 0.4 MeV threshold, the estimated beta energy of the mixture, approximately 0.440 MeV, justifies the use of 0.5 for the ϵ_s.

The beta minimum detectable concentrations (MDC) were 330 and 300 dpm/100 cm^2. The physical surface area assessed by the gas proportional detectors were 126 cm^2.

Removable Activity Measurements

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

[2] International Standard. ISO 7503-1, Evaluation of Surface Contamination - Part 1: Beta-emiters (maximum beta energy greater than 0.15 MeV) and alpha-emitters. August 1, 1988.

RADIOLOGICAL ANALYSIS

Gross Alpha/Beta

Smears were counted on a low-background gas proportional system for gross alpha and beta activity. The MDCs of the procedure were 8 dpm/100 cm^2 and 15 dpm/100 cm^2 for gross alpha and gross beta, respectively.

DETECTION LIMITS

Detection limits, referred to as minimum detectable concentration (MDC), were based on 3 plus 4.65 times the standard deviation of the background count [$3 + (4.65\sqrt{BKG})$]. When the activity was determined to be less than the MDC of the measurement procedure, the result was reported as less than MDC. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument.