UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

May 23, 2003

Mr. 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 50-344/2003-002

Dear Mr. Quennoz:

An NRC inspection was conducted on April 21-24, 2003, at the Trojan Nuclear Plant. This 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 consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel. Enclosure 1 presents the results of this inspection. Overall, the inspection found that decommissioning activities were being conducted in accordance with procedural and regulatory requirements.

During the inspection, individuals from the Oak Ridge Institute for Science and Education (ORISE), Environmental Survey and Site Assessment Program, conducted embedded piping surveys and collected sediment and water samples on behalf of the NRC. The preliminary NRC conclusions about this portion of the inspection are presented in Section 4 of the enclosed NRC inspection report. The staff's evaluation of the comparative survey results and a copy of the ORISE report will be transmitted to you under separate correspondence at a later date.

During the inspection period, the NRC conducted an in-office review of changes to the Physical Security Plan to determine if the changes decreased the effectiveness of the Plan and to ensure that the requirements of 10 CFR 50.54(p) were met. In summary, no findings of significance were identified. Details of this review are provided in Enclosure 2.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) will be made 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/reading-rm/adams.html (the Public Electronic Reading Room).

If you have any questions concerning this inspection, please contact Mr. Robert J. Evans, Senior Health Physicist, at (817) 860-8234 or the undersigned at (817) 860-8191.

Sincerely,

/RA/ E. E. Collins for

D. Blair Spitzberg, Ph.D., Chief
Fuel Cycle and Decommissioning Branch

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

Enclosures:
1. NRC Inspection Report
050-00344/2003-02
2. Physical Security Plan Review

cc w/enclosures:
Lansing G. Dusek
Manager, Plant Support
Portland General Electric Company
Trojan Nuclear Plant
71760 Columbia River Highway
Rainier, Oregon 97048

Chairman
Board of County Commissioners
Columbia County
St. Helens, Oregon 97501

David Stewart-Smith
Oregon Office of Energy
625 Marion Street NE
Salem, Oregon 97301

Lloyd K. Marbet
19142 S.E. Bakers Ferry Road
Boring, Oregon 97009

Jerry Wilson
Do It Yourself Committee
570 N.E. 53rd
Hillsboro, Oregon 97124

Eugene Roselie
Northwest Environment Advocates
133 S.W. 2nd Avenue
Portland, Oregon 97204

Douglas Nichols, Esq.
Vice President, General Counsel and Secretary
Portland General Electric Company
121 SW Salmon Street
Portland, Oregon 97204

Michael B. Lackey
General Manager, Trojan
Portland General Electric Company
Trojan Nuclear Plant
71760 Columbia River Highway
Rainier, Oregon 97048

bcc w/enclosure (via ADAMS distrib):
EECollins
DBSpitzberg
JBuckley, NMSS/DWM/DCB
JCDehmel, NMSS/DWM/DCB
CMCraig, NMSS/DWM/DCB
GAPick, PSB/DRS
TWPruett, PSB/DRS
RJEvans
EMGarcia
NBHolbrook
FCDB File

ADAMS: ☒Yes ☐No Initials: RJE

☒Publicly Available ☐Non-Publicly Available ☐Sensitive ☒Non-Sensitive

DOCUMENT NAME: S:DNMS\Fcdb\RJE\tn0302.wpd Final: R:_tn\2003\tn0302.wpd

RIV:DNMS:FCDB	NMSS:DWM:DCB	NMSS:DWM:DCB
RJEvans	JCDehmel	CMCraig
/RA/	***/RA/* via e-mail RJEvans for**	***via e-mail VEverett for***
05/22/03	05/23/03	05/23/03

DRS:PSB	C:PSB	C:FCDB
GAPick	TWPruett	DBSpitzberg
/RA/ TWPruett for	***/RA/***	***/RA/ DBSpitzberg for***
05/23/03	05/23/03	05/23/03

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-344
License No.:	NPF-1
Report No.:	50-344/2003-002
Licensee:	Portland General Electric Company
Facility:	Trojan Nuclear Plant
Location:	71760 Columbia River Highway Rainier, Oregon 97048
Dates:	April 21-24, 2003
Inspectors:	Robert J. Evans, PE, CHP, Senior Health Physicist Fuel Cycle & Decommissioning Branch Jean-Claude F. Dehmel, CHP, Health Physicist Decommissioning Branch Division of Waste Management Office of Nuclear Material Safety and Safeguards
Accompanied By:	Ernesto Quinones-Padovani, Physical Scientist High Level Waste Branch Division of Waste Management Office of Nuclear Material Safety and Safeguards
Approved by:	D. Blair Spitzberg, Ph.D., Chief Fuel Cycle & Decommissioning Branch
Attachment:	Supplemental Information
ADAMS Entry:	IR 05000344-03-02; on 04/21-24/2003; Portland General Electric Co.; Trojan Nuclear Plant; Decommissioning Report; No violations.

EXECUTIVE SUMMARY

Trojan Nuclear Plant
NRC Inspection Report 50-344/2003-02

The inspectors reviewed the licensee's implementation of the self-assessment, audits and corrective action programs; maintenance and surveillance activities; decommissioning activities; and the final survey program. The inspectors also conducted a followup review of three previous NRC inspection findings. In summary, the licensee was conducting decommissioning activities in accordance with regulatory and license requirements.

Self-Assessment, Auditing, and Corrective Action

- The licensee's quality assurance audit program met 10 CFR Part 50, Appendix B, requirements. The Independent Review and Audit Committee was functioning as stipulated in the Permanently Defueled Technical Specifications. Radiological incidents were being documented and dispositioned. Corrective action request reports were being used to document quality-related concerns. An NRC Inspection Followup Item was created to ensure that a potentially negative trend involving procedure compliance has been reviewed and corrected by the licensee (Section 1).

Maintenance and Surveillance

- The licensee conducted maintenance and surveillance activities in accordance with procedural requirements. The inspectors questioned the licensee's control of fire protection procedures, and as a result, the licensee issued a corrective action request to ensure continued compliance with National Fire Protection Association code requirements (Section 2).

Decommissioning Performance and Status Review

- Decommissioning activities were being conducted in accordance with license and regulatory requirements. Radiation protection controls and plant systems were in place to support decommissioning activities. The operations shift crew composition met technical specifications requirements (Section 3.2.a).

- The licensee was maintaining the spent fuel pool within technical specifications requirements for water level, boron concentration, and water temperature. The licensee maintained sources of makeup water for the spent fuel pool. During the inspection, the licensee experienced a loss of spent fuel pool indications when a power supply was accidently cut. The control room maintained positive oversight of the spent fuel pool via alternate means. A corrective action request was issued by the licensee to evaluate the incident (Section 3.2.b).

Inspection of Final Surveys

- Comparative surveys were conducted by both the licensee and representatives from the Oak Ridge Institute for Science and Education (ORISE) on embedded piping in the auxiliary building. Differences were identified between the licensee's and NRC's sample results. The resolution of these differences and the interpretation of the survey measurements will be evaluated once ORISE issues its report to the NRC. This issue is considered an Inspection Followup Item (Section 4.2.a).

- One sediment and two water samples were collected by the NRC during the inspection. The sample results will be evaluated and reported to the licensee under separate correspondence at a later date (Section 4.2.b).

Followup

- Three previously identified NRC Inspection Followup Items were reviewed. Two of three items were closed (Section 5).

Report Details

Summary of Facility Status

The Trojan Nuclear Plant was permanently shutdown during January 1993. The major dismantlement activities have been completed at the site. At the time of this inspection, work in progress included decommissioning and decontamination of the auxiliary and fuel buildings. Final status surveys and confirmatory surveys were in progress in the turbine building. Operational surveys were being conducted in the auxiliary and fuel buildings. The main steam support structure was scheduled to be final status surveyed in the near future.

The licensee started moving fuel from the spent fuel pool (SFP) to multi-purpose canisters on December 31, 2002. At the conclusion of the onsite inspection, 15 concrete casks, each containing a fully loaded canister, had been moved to the Independent Spent Fuel Storage Installation (ISFSI) pad. The licensee planned to load the remaining fuel into 19 more canisters. Fuel loading was scheduled to be completed during late-October 2003.

1 Self-Assessment, Auditing, and Corrective Action at Permanently Shutdown Reactors (40801)

1.1 Inspection Scope

The inspectors evaluated the effectiveness of the licensee's ability to identify, resolve, and prevent issues that had the potential to degrade safety or the quality of decommissioning. The inspectors reviewed the licensee's self-assessment, program audit, corrective action report, and root cause evaluation controls.

1.2 Observation and Findings

Appendix B of 10 CFR Part 50 states that the licensee shall be responsible for the establishment and execution of a quality assurance (QA) program. Further, this program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions. The inspectors reviewed the licensee's implementation of its QA program. In particular, the inspectors reviewed the QA audit of 10 CFR Part 50 activities. This audit was concluded on April 17, 2003. The audit identified four strengths, no findings, seven observations, and two recommendations. Overall, the inspectors concluded that the audit was conducted in accordance with regulatory and license requirements.

Permanently Defueled Technical Specifications (PDTS) 5.5.2 states that the Independent Review and Audit Committee (IRAC) is responsible for reviewing and advising the Trojan site executive on matters relating to safe storage of irradiated fuel. This section of PDTS also describes IRAC composition, meeting frequency, quorum requirements, review responsibilities, and recordkeeping requirements. In addition, Trojan Plant Procedure TPP 10-8 described the IRAC and the ISFSI Safety Review Committee.

The inspectors reviewed the licensee's implementation of PDTS 5.5.2 and Procedure TPP 10-8 requirements. The inspectors' review included the committee's composition, meeting frequency, quorum, and record handling. Meeting minutes reviewed included routine quarterly Meeting No. 2003-01 conducted on March 18, 2003, and special Meeting No.2003-13 conducted on April 16, 2003. In summary, the records indicate the licensee's compliance with PDTS and procedural requirements. No unusual discrepancies were found during the review of the licensee's IRAC documentation.

The 2002-2003 radiological event reports and personnel/clothing contamination logs were reviewed to determine if any negative trends existed in the radiation protection program area. At the time of this inspection, there were eight radiological event reports for 2002-2003. The events included loss of control of contaminated equipment in the restricted area, unintentional transfer of potentially contaminated water through floor drains and embedded piping, discovery of contamination in an unrestricted area, and erroneously high neutron dose rate measurements due to instrument error. All events appeared isolated although three events were related to the unintentional transfer of potentially contaminated water through abandoned pipe penetrations and floor drains. In each instance, the licensee conducted an evaluation of the apparent cause of the incident, documented the immediate corrective actions taken, and listed the longer term corrective actions required to prevent recurrence. Overall, the licensee was using the radiological event report process to identify and correct problems that involved radioactive material.

The 2002-2003 personnel/clothing contamination logs were reviewed to determine if recent contamination events were repetitive in nature. The event with the highest measured level of contamination involved the discovery of contamination on a worker that was lifting and moving potentially contaminated equipment. In this case, the licensee conducted a dose assessment of the clothing and skin contamination and determined that the worker did not receive a skin dose above the action level. The calculated skin dose was 240 millirads with an action level of 750 millirads. In summary, the licensee was using the contamination logs to keep track of worker contamination incidents, and no obvious trend was identified during the review of these logs.

The inspectors also reviewed the licensee's corrective action request log for 2003 to determine if quality-related problems were being identified and dispositioned. During January-April 2003, 17 corrective action requests were written. Of that number, seven were related to apparent procedural non-compliances involving fuel handling activities. Although none of the seven procedural non-compliances were safety significant, collectively, the number of non-compliances suggested that an adverse trend existed.

In response to this apparent adverse trend, the licensee conducted a generic root cause evaluation during February 2003 which included proposed corrective actions for preventing recurrence. Unfortunately, another procedure non-compliance incident occurred during March 2003 suggesting that the generic corrective actions were not entirely successful. This subject area is considered an Inspection Followup Item (50-344/0302-01) to ensure that the licensee's corrective action program has implemented effective actions to eliminate this apparent negative trend.

1.3 Conclusion

The licensee's QA audit program met 10 CFR Part 50, Appendix B, requirements. The IRAC was functioning as stipulated in PDTS. Radiological incidents were being documented and dispositioned. Corrective action request reports were being used to document quality-related concerns. An NRC Inspection Followup Item was created to ensure that a potentially negative trend involving procedure compliance been reviewed and corrected by the licensee.

2 Maintenance and Surveillance at Permanently Shutdown Reactors (62801)

2.1 Inspection Scope

The inspectors observed selected maintenance and surveillance activities to verify if site structures, systems, and components were being maintained in compliance with PDTS and procedural requirements. The inspectors observed monthly surveillance testing of the area radiation monitors, monthly preventive maintenance inspection of the fuel building crane, and periodic performance testing of the diesel fire pump.

2.2 Observation and Findings

a. Area Radiation Monitor Channel Functional Tests

Two area radiation monitors were still in service. Monitors RIS-6112 and RIS-6113 were located near the SFP and serve as criticality alarms in the spent fuel handling and storage areas. The inspectors observed the licensee conducting the monthly channel functional test of the two monitors. The work was conducted using instructions provided in Periodic Operating Test POT 26-1, "Area Radiation Monitoring System," Revision 22.

During the surveillance, the area radiation alarm setpoints were compared to the requirements listed in Section 5.6.1.3.1 of the Defueled Safety Analysis Report (DSAR). The DSAR states that with fuel in the SFP or fuel building, the area radiation monitors shall be operable with their alarm/trip setpoints set at or less than 15 millirems per hour. The inspectors noted that the as-left alarm setpoints were below the maximum value specified in the DSAR. The alert alarm was set at 2.5 millirems per hour, while the high alarm was set at 9 millirems per hour.

In summary, the two monitors passed the channel checks, the alarm setpoints were more conservative than DSAR requirements, and all alarms and indications worked as designed.

b. Fuel Building Crane Monthly Preventive Maintenance

A monthly preventive maintenance inspection of the fuel building crane was conducted. The maintenance work included visual inspection of the hoist ropes, rope reeving, hoist attachment, gear boxes, and remainder of crane components. The licensee then operated the crane and its two hoists through the limits of travel to ensure crane and

hoist movement in all directions. The preventive maintenance work was completed in accordance with Maintenance Procedure MP 1-20, "Cranes, Hoists, and Winches," Revision 16, and Trojan Plant Procedure TPP 14-25, "Hoisting Equipment Control," Revision 0. The crane passed the preventive maintenance inspection with no problems encountered.

c. Periodic Testing of Diesel Fire Pump

A fire protection program was developed to ensure that appropriate fire protection measures are maintained to protect the facility from fires which could impact the safe storage of irradiated fuel or the release of radioactive materials. Details of the fire protection program are provided in the Trojan Nuclear Plant Fire Protection Plan, PGE-1012, Amendment 26.

Trojan currently has two redundant fire pumps, one electric and one diesel. Table 3-1 of the fire protection plan states that the electric and diesel fire pumps are tested monthly. During the inspection, the licensee conducted a monthly test of the diesel fire pump. The work was conducted using guidance provided in Work Order 31802 and Periodic Operating Test POT 10-2-DB, "Diesel Fire Pump," dated January 29, 2003. All operating parameters (engine speed, pressures, temperatures, etc.) were within procedural limitations. The diesel fire pump subsequently passed the monthly test and was returned to service.

During the review of the diesel fire pump test documentation, the inspectors questioned the licensee's control of the fire protection procedures. The inspectors noted that the licensee no longer controlled the procedures per PDTS 5.7.1 requirements. The licensee claimed that fire protection system procedures were no longer required to be controlled in accordance with PDTS because the fire hazards analysis demonstrated that there is no credible fire event that could impact the safe storage of fuel in the SFP. In other words, since the fire hazards analysis concluded that there was no credible fire event, the licensee downgraded the diesel fire pump surveillance procedures (and similar electric driven fire pump procedures) from PDTS controlled procedures to maintenance work instructions.

However, during the inspection, the licensee determined that it may have made changes to the fire protection program without ensuring compliance with National Fire Protection Association code requirements. The licensee committed in its Trojan Nuclear Plant Fire Protection Plan to meet code requirements. When the licensee downgraded the fire protection program procedures, the licensee also stopped screening fire protection procedure changes against current code requirements. The licensee issued a corrective action request to determine if the current program was still in compliance with code requirements.

2.3 Conclusions

The licensee conducted maintenance and surveillance activities in accordance with procedural requirements. The inspectors questioned the licensee's control of fire protection procedures, and as a result, the licensee issued a corrective action request to

ensure continued compliance with National Fire Protection Association code requirements.

3 Decommissioning Performance and Status Review at Permanently Shutdown Reactors (71801)

3.1 Inspection Scope

The inspectors reviewed the status of decommissioning and evaluated whether the licensee and its contracted workforce were conducting decommissioning activities in accordance with license and regulatory requirements.

3.2 Observation and Findings

a. Plant Tours

Plant tours were conducted to observe decommissioning and operational activities in progress. Decontamination activities included remediation of potentially contaminated concrete. The work was being conducted in a safe and orderly manner. Radiological controls, including postings and barriers, were in place as needed in the areas where decommissioning activities were in progress. The inspectors noted good housekeeping and fire protection practices. The inspectors also conducted radiological surveys while in the restricted area using a Ludlum Model 2401-EC2 survey meter calibrated to cesium-137 (NRC No. 016295G). All areas within the restricted area with elevated radiation levels had been identified and posted by the licensee, including high radiation areas.

Technical Specification Table 5.2.2-1 lists the minimum shift crew composition. During plant tours, the inspectors made unannounced visits to the control room. Control room and plant operations staffing met the minimum crew composition as specified in PDTS.

b. Spent Fuel Pool Safety

Permanently Defueled Technical Specifications 3.1.1 requires that SFP water level be greater than or equal to 23 feet over the top of irradiated fuel assemblies seated in the storage racks. During the inspection, the pool level was 24 feet, 8 inches. Technical Specifications 3.1.2 requires that the SFP boron concentration be greater than or equal to 2000 parts per million. As of April 21, 2003, the boron concentration was 2196 parts per million. Technical Specifications 3.1.3 specifies that SFP coolant temperature be maintained less than or equal to 140 degrees Fahrenheit. The coolant temperature was 90 degrees during the inspection. In summary, the licensee was maintaining the SFP in accordance with PDTS requirements.

The inspectors conducted a review of SFP support systems. Train A of the modular SFP cooling and cleanup system was in service with Train B in standby. The licensee recently placed the Train B air cooler and fan assembly in standby because of the reduced heat load in the pool. In addition, the SFP water makeup sources were

reviewed. The normal makeup was domestic water. Alternate sources included service water, fire protection, and gasoline-powered portable pumps. All water sources were available during the inspection.

On April 23, 2003, the licensee experienced a loss of power event that resulted in a loss of control room indicators for the SFP. The power supply for the SFP temperature indicator, high temperature alarm, and water level high/low alarm in the control room was accidently severed by decommissioning contractors. A power supply cable was accidently cut by contractors in the cable spreading room. Upon loss of the SFP indicators, the control room operators implemented an off-normal response procedure. A maintenance request was issued to repair the severed cable. The following day, the control room indicator and alarms were returned to service when the cable was repaired.

During the loss of instrument power incident, the licensee maintained positive control over the SFP via alternate means. A temporary temperature indicator was installed, and the SFP water level was continuously monitored via a remote controlled video camera. Further, the modular SFP cooling and cleanup system skid instrumentation included non-calibrated temperature indicators that provided a relatively accurate indication of SFP coolant temperature.

The licensee issued a corrective action request report to investigate the cause of the loss of SFP indication event. The evaluation was incomplete at the end of the inspection period but was expected to include the cause of the incident and proposed corrective actions to prevent recurrence of the event.

3.3 Conclusions

Decommissioning activities were being conducted in accordance with license and regulatory requirements. Radiation protection controls and plant systems were in place to support decommissioning activities. The operations shift crew composition met PDTS requirements.

The licensee was maintaining the SFP within the PDTS requirements for water level, boron concentration, and water temperature. The licensee maintained sources of makeup water for the SFP. During the inspection, the licensee experienced a loss of SFP indications when a power supply was accidently cut. The control room maintained positive oversight of the SFP via alternate means. A corrective action request was issued by the licensee to evaluate the incident.

4 Inspection of Final Surveys at Permanently Shutdown Reactors (83801)

4.1 Inspection Scope

The inspectors reviewed licensee activities associated with the final status survey to determine compliance with the Decommissioning Plan and License Termination Plan requirements. A series of comparison surveys was conducted on embedded piping to assess the adequacy and accuracy of the licensee's final status survey methods. Also,

water and sediment samples were taken in the pipe penetration area to determine the presence of the plant-derived radioactivity in groundwater.

4.2 Observations and Findings

a. Comparative Surveys of Embedded Piping Located in the Auxiliary Building

During the inspection, representatives from the Oak Ridge Institute for Science and Education (ORISE), under contract with the NRC, conducted a series of surveys on embedded drain piping located in the auxiliary building. The piping consisted of the former dirty radwaste system piping. The purpose of the surveys was to compare survey results generated by the licensee as a preparatory step leading to the conduct of NRC confirmatory surveys and issuance of final status survey reports for these systems. The major objectives of the surveys were to determine residual levels of activity in piping after remediation, compare these results with those of the licensee, identify any differences in survey measurements between survey techniques and instrumentation used by ORISE and the licensee, and conduct a preliminary evaluation of the results against derived concentration guideline level (DCGL) established for embedded piping.

A survey plan was developed by NRC/ORISE following a review of the licensee's draft survey packages, walkdown of the survey units, and discussions with the licensee's staff. The survey methodology consisted of internal surface scans using beta particulate radiation detectors and measurements taken at fixed locations along the piping. The survey equipment consisted of a ratemeter/datalogger (Ludlum Model 2221) connected to an array of four Geiger-Mueller pancake detectors (Ludlum 44-9), mounted at 90^{o} intervals on a single shuttle with each detector connected to its own ratemeter. The detector array was pulled through the embedded piping and readings were observed and recorded as the shuttle traveled through the length of the piping. In addition, a series of fixed measurements were made at specific locations using scan readings to identify the areas with the highest activity levels. For some portions of the embedded piping, visual inspections were made using the licensee's video monitoring system to assess the internal conditions of the piping.

Three segments of the embedded drain piping of the former dirty radwaste system were evaluated as part of this comparative survey. The portions of the system were designated as survey units No. P61203A, P61203B, and P61203C, comprising about 40 m^2 of piping. Observations noted during the conduct of the surveys revealed that the segments of the piping were free of loose residues and standing water.

A review of preliminary survey results indicated that the residual radioactivity, when noted, was typically spotty and isolated over short sections of piping, and that most measurements were found to be non-distinguishable from background radiation levels. However, a preliminary review of side-by-side measurements taken at six locations with elevated residual activity levels indicated significant differences among some of the licensee's results, at times up to several orders of magnitude. Among other reasons, the difference in results may be attributed to the types of survey instrumentation used by ORISE (array of four beta detectors) and the licensee (single sodium iodide detector), calibration standards and methods, and survey techniques. The resolution of these

differences and interpretation of survey measurements will be evaluated by the staff once ORISE issues its report to the NRC. This issue will be tracked as Inspection Followup Item 50-344/0302-02.

b. Sample Collection in the Pipe Penetration Area of the Auxiliary Building

As a separate task assigned by the NRC, the ORISE team, with assistance from the licensee's staff, collected two water samples and one sediment sample in the pipe penetration area of the auxiliary building. The samples will be evaluated for the presence of plant-derived radioactivity, including tritium and hard-to-detect radionuclides. The licensee collected its own set of duplicate samples at the same time for comparison after ORISE issued its laboratory analysis report. The analysis of the samples will be conducted by ORISE using a high resolution gamma spectroscopy (EG&G/Ortec system) and an alpha spectroscopy system (Canberra system). For tritium, the analysis will be conducted using a liquid scintillation counting system (Packard A3100 system). For the sediment sample, the results will be expressed in terms of concentrations (pCi/g-dry) for all identified radionuclides and compared to NRC screening and/or site-specific DCGLs for soils. For water samples, the results will be reported as pCi/L and compared to the U.S. Environmental Protection Agency's Maximum Contaminant Levels for ground water.

The results of the sediment and water samples will be evaluated by the NRC after ORISE has analyzed and reported all of the results to the NRC staff. The staff's evaluation of sample results and a copy of the ORISE laboratory report will be transmitted to the licensee under separate correspondence at a later date.

4.3 Conclusions

Comparative surveys were conducted by both the licensee and representatives from ORISE on embedded piping in the auxiliary building. Differences were identified between the licensee's and NRC's sample results. The resolution of these differences and the interpretation of the survey measurements will be evaluated once ORISE issues its report to the NRC. This issue is considered an Inspection Followup Item.

One sediment and two water samples were collected by the NRC during the inspection. The sample results will be evaluated and reported to the licensee under separate correspondence at a later date.

5 Followup (92701)

5 .1 (Closed) Inspection Followup Item 50-344/0003-07: Scan & Static MDCs and Instrument & Surface Efficiencies

At the time of the August 14-17, 2000, inspection, the licensee had not completed the calculations for scan and static minimum detectable concentrations (MDCs) pending the receipt of appropriate calibration sources. At that time, the NRC was unable to review the licensee's methodology for calculating the scan and static MDCs, the methodology

for calculating the weighted beta energy instrument efficiency (ε_i), and the technical basis for determining surface efficiency (ε_s) for measuring contamination over irregular surfaces such as scabbled concrete.

In a followup inspection conducted during May 13-16, 2002, the licensee's staff revealed that arrangements had been made with Oregon State University (OSU) to empirically develop surface efficiencies using concrete surfaces expected to be found at the Trojan plant during the conduct of the final status surveys. The licensee supplied OSU with a number of concrete samples with different surfaces, i.e., scabbled using different tools. A review of the work plan indicated that the project milestones addressed the following relevant issues: define the profile of concrete surface surfaces, apply known amounts of radioactive contamination on prepared concrete surfaces, make beta radiation measurements, establish surface and detector efficiencies, and prepare a technical basis document.

During this inspection, the licensee indicated that the OSU report[1] had been finalized and evaluated for its application to the Trojan site. The application of the OSU report is contained in the PGE document: "Final Survey Technical Basis Document (TBD) Site Specific Scabbled Concrete Source Efficiency Factor, FSC 2003-03, 4/21/03, Revision 0." The TBD uses only the OSU results for concrete surfaces doped with technetium-99 and supplements the OSU data with its own set of measurements, all independently made by licensee staff. As a radionuclide, technetium-99 was deemed to be representative of the weighted average beta particle energy for the expected mix of radioactive contaminants.

The licensee's staff used two types of radiation detectors to represent survey methods in use at the plant. The detectors were gas-flow proportional detectors, Model No. 43-68 and 44-116, both made by Ludlum Measurements, Inc. In its evaluation, the licensee assumed that the detector efficiency was 0.5, based on a 2-pi geometry for the control matrix. A review of the TBD indicated that surface and detector efficiencies were derived for five types of scabbled concrete surfaces, in addition to defining control and reference surfaces. The concrete surfaces were prepared using five types of tools, including a spade bit, bush head, one "Fingered Jack," needle gun, and floor scabbler. In terms of roughness, the spade bit generated the most uneven surfaces, while the floor scabbler generated smoother surfaces. In its conclusions, the licensee determined that single source efficiency of 0.3 can be conservatively assigned for any type of scabbled concrete, given that the results ranged from 0.395 to 0.507. The TBD acknowledged that surface efficiency was affected by competing factors, including surface roughness (expressed as effective height in the TBD), surface area under the detector, source-to-detector distance, self-absorption, and backscatter. The TBD and the licensee's assumption leading to the selection of a value of 0.3 for a generic surface efficiency were deemed acceptable to the NRC inspectors. However, the inspectors recommended that the licensee determine the detector efficiency (ε_i) empirically as this would make the results and conclusions of the study more robust and defensible. The

[1] Research to Determine Source Efficiencies (ε_s) for Scabbled and Rough Concrete Surfaces, Mr. Michael T. Bak, Master of Science Thesis, Oregon State University, March 18, 2003.

detector efficiency could be determined using a National Institute of Standards and Technology traceable standard certified for a 2-pi emission rate, or submit the control matrix used in the study to a laboratory for certification as a secondary standard.

At this time, this Inspection Followup Item is considered closed; however, the implementation of the derived surface efficiencies (ε_s) in converting raw instrument readings to results directly comparable to the DCGL will be the subject of future NRC inspections and will be reviewed as part of the review process of final status survey reports submitted by the licensee.

5.2 (Closed) Inspection Followup Item 50-344/0003-11: Embedded Piping Surveys

Based on the observations of a prior inspection, it was determined that the licensee's final status survey program for embedded piping and other difficult to access or inaccessible areas had not been finalized. The licensee previously estimated that about 6,000 feet of embedded piping will remain in the auxiliary building and about 4,000 feet will remain in the fuel building. At this time, the target decontamination levels for embedded piping are less than 1,000 dpm/100 cm^2 for removable activity, less than 7,250 dpm/100 cm^2 for the average amount of fixed activity, and less than 50,000 dpm/100 cm^2 for the maximum amount of fixed activity. An evaluation of the proposed DCGL for embedded piping was found to be acceptable - see NRC Inspection Report 50-344/2002-004. However, it was noted that its implementation, under some circumstances, may require careful evaluation especially for survey units with results approaching the DCGLs for both the room itself and any associated embedded pipe present in walls, floor, or ceiling. It was also noted that the evaluation will need to ensure that, under all circumstances, the sum of the doses from all sources of residual radioactivity does not exceed the annual dose limit of 25 mrem (0.25 mSv) in any portion of the survey unit.

During this inspection, it was determined that the licensee had revised specific procedures to address these issues. The procedures that were revised included Radiation Protection Procedures RP-459 (Final Survey Instruments Calibration Program, 4/10/03, Revision 3); RP-453 (Final Survey Data Processing, 11/25/02, Revision 3); RP- 456 (Control and Operation of Data Logger Survey Instruments, 4/10/03, Revision 4); RP-452 (Final Survey data Collection, 4/10/03, Revision 8); Trojan Plant Procedure TPP 14-24 (Transfer of Turnover Units, 2/25/03, Revision 3); and instructions noted in the Final Survey Unit Design Specifications package (Attachment 1 to RP-451, Final Survey Unit Design, 11/6/02, Revision 5). The licensee has agreed to revise another procedure (RG 20-6, Embedded Piping Final Survey, 11/5/98, Revision 0) to ensure that it is consistent with recently revised procedures and stated objectives.

In addition to these revisions, the licensee has agreed to implement the criteria identified in the associated calculation package, "*Dose Model for Embedded Piping, RPC 2000-08, 10/11/00, Revision 0*." The most important ones specify that more rigorous shielding calculations be conducted whenever external exposure rates approach the 5 mrem/year criterion; both sides of any shield wall need to be evaluated to confirm that the dose criterion is met on either side; particular attention is required for

areas with the highest pipe density and thicknesses of concrete less than 5 inches; and any piping with levels approaching the DCGL of 100,000 dpm/100 cm^2 will be evaluated for further actions, i.e., shielding analysis using area-specific conditions, elevated measurement provisions, or subjected to further remediation.

The licensee's radiation detection system has been built by Ludlum Measurements, Inc. using specifications developed by the licensee. At this time, the specifications of the system are still considered proprietary; however, the staff was able to review detector diagrams and discuss with licensee staff the important operating features of the system. The licensee has procured five radioactive standards from Isotopes Products Laboratory specifically designed to address the calibration of the gamma radiation detector used in surveying embedded piping. The standards include three cobalt-60 and two cesium-137 calibration sources, provided with National Institute of Standards and Technology traceable certificates. In addition, the Survey Data Management System functions have been updated to include codes for the types of measurements and detectors used in surveying embedded piping. The codes include provisions for static and scan measurements, removable activity, and for the two types of gamma radiation detectors (sodium and cesium iodide) that will be used for such surveys.

Finally, the licensee demonstrated the use of a video monitoring system to confirm that the piping is free of loose residues and standing water before initiating final status surveys. The licensee indicated that the grouting of embedded piping will be considered only after the piping has been shown to be in compliance with the DCGL and that the NRC has been given an opportunity to conduct its own confirmatory surveys.

At this time, this Inspection Followup Item is considered closed; however, the implementation of the survey program for embedded piping will be the subject of the NRC's evaluation of comparative survey results obtained during this inspection, any future NRC inspections, and as part of the review process of final status survey reports submitted by the licensee.

5.3 (Open) Inspection Followup Item 50-344/0202-05: Evaluation of Site-Specific DCGLs Derived for Soils

A review of two site-specific DCGLs was conducted, based on derivations documented in the following calculation packages: "*Gross Activity DCGL for Systems, Structures, and Components Impacted Primarily Due to Primary to Secondary System Leaks, RPC 2002-06, approved October 9, 2002*," and "*Gross Activity DCGL for the Fuel Building, FSC 2009-09, approved December 24, 2002*." The licensee indicated that the DCGL calculation package (RPC 2002-06) addressing primary-to-secondary system leaks will be used for all building exterior surfaces. The derivation was found to be based on the methodology presented in the Decommissioning Plan and License Termination Plan (Section 4.2.3.2), information obtained from NUREG-1507, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions, and information referencing licensee communications to the NRC. The NRC's evaluation confirmed the licensee's site-specific DCGL values and its implementation, but identified a number of items requiring technical clarifications.

The DCGLs are based on screening DCGLs developed for the building occupancy scenario, with radionuclide specific DCGLs listed in Table 4-1 of the Decommissioning Plan and License Termination Plan. The DCGLs reflect radionuclide distribution and fractions using samples believed to be representative of radiological conditions within buildings or systems expected at the conclusion of remediation activities. The samples were taken at biased locations where plant-derived radioactivity and transuranic (TRU) and hard-to-detect (HTD) radionuclides would be readily detected, if present. The samples consist of concrete, metal filings, pipe debris, paint/grout, and spent resins. The samples were analyzed by Duke Engineering and Services Environmental Laboratory and Thermo NUTech under contract to the licensee. A review of the results of the ten samples taken in building areas or systems indicated the presence of major contaminants, including tritium, iron-55, cobalt-60, nickel-63, strontium-90, niobium-94, antimony-125, cesium-134 and cesium-137. Also, alpha emitting radionuclides were reported to be above analytical MDCs, reported as plutonium-238/239/240, americium-241, and curium-242/243/244. A review of the results for TRU and HTD radionuclides indicated that the reported concentrations were within or near their respective analytical MDCs, ranging from about 0.05 to 1.5 pCi/g for TRUs and from about 0.1 to 29 pCi/g for HTDs. The reported laboratory MDC levels were found to be generally adequate based on a comparison with the screening DCGLs adopted by Table 4-1. In their implementation, final status survey results will be evaluated using the applicable DCGL using procedure RP-453 (Final Survey Data Processing) and instructions given in the Final Survey Unit Design Specifications package. In turn, the Final Survey Unit Design Specifications package refers to procedure RP-457 (Development and Application of Derived Concentration Guideline Levels) which provides further details on their applications.

A review of the information presented in each calculation package indicated that further technical clarifications were needed. The technical issues noted include describing the basis or process as to how two spent resin samples were composited (samples No. T91-6 and TB-SP-001); addressing the rationale for excluding the presence of TRUs that were reported above analytical detection limits (samples No. MS-SC-02, FB-45-014, FB-77-005, and TB-SP-001); indicating that the DCGL for fuel building is only for specific areas where the presence of TRUs has not been detected nor is expected; and stating that a separate DCGL will be developed for areas of the fuel building known or expected to have been contaminated with TRUs. These concerns were addressed by the licensee in the revised versions of both calculation packages, both reissued on May 7, 2003.

The implementation of the DCGLs in survey areas or units will be the subject of future NRC inspections, and as part of the review process of final status survey reports submitted by the licensee. The evaluation of the basis of the DCGL for areas of the fuel building and SFP cooling and cleanup system known or expected to be contaminated with TRUs will be conducted once the calculation package has been prepared by the licensee and made available to the NRC.

6 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management and staff at the exit meeting on April 24, 2003. During this inspection, the licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors. The review of the information describing the proprietary design of the radiation detectors used for embedded piping survey was not recorded in NRC inspection notes, nor was any related information taken by the staff.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Portland General Electric

J. Cooper, Radiation Protection
K. Cox, ISFSI Manager
L. Dusek, Plant Support Manager
S. Ford, Licensing Engineer
G. Huey, Radiation Protection Tech Support Manager
M. Lakey, General Manager
T. Meek, Radiation Protection Manager
J. Mihelich, Engineering Manager
S. Nichols, Decommissioning Projects Manager
S. Schneider, Operations Manager
J. Vingerud, Maintenance Manager
J. Westvold, Nuclear Oversight Manager

Oak Ridge Institute for Science and Education

L. Mashburn, Health Physics Technician
T. Vitkus, Survey Projects Manager

State of Oregon

A. Bless, Oregon Office of Energy, Trojan Resident Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-344/0302-01	IFI	Followup of Corrective Actions Taken for Apparent Adverse Trend
50-344/0302-02	IFI	Discrepancies with Comparative Surveys of Embedded Piping

Closed

50-344/0003-07	IFI	Scan & Static MDCs and Instrument & Surface Efficiencies
50-344/0003-11	IFI	Embedded Piping Surveys

Discussed

50-344/0202-05	IFI	Evaluation of Site-Specific Derived DCGLs for Soils

LIST OF ACRONYMS

CFR	Code of Federal Regulations
DSAR	Defueled Safety Analysis Report
DCGL	derived concentration guideline level
dpm/100 cm^2	disintegrations per minute per 100-square centimeters
ε_i	instrument efficiency
ε_s	surface efficiency
HTD	hard-to-detect (radionuclides)
IFI	Inspection Followup Item
IRAC	Independent Review and Audit Committee
ISFSI	Independent Spent Fuel Storage Installation
m^2	square meters
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual (NUREG-1575)
MDC	minimum detectable concentrations
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
OSU	Oregon State University
PDTS	Permanently Defueled Technical Specifications
pCi/g	picocuries per gram
QA	quality assurance
SFP	Spent Fuel Pool
TBD	Technical Basis Document
TRU	transuranics

ENCLOSURE 2

PHYSICAL SECURITY PLAN REVIEW

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-344
License No.:	NPF-1
Report No.:	50-344/03-02
Licensee:	Portland General Electric Company
Facility:	Trojan Nuclear Plant
Location:	71760 Columbia River Highway Rainier, Oregon 97048
Date:	April 23, 2003
Inspector:	Gregory A. Pick, Senior Physical Security Inspector Plant Support Branch Division of Reactor Safety
Approved By:	Troy W. Pruett, Chief Plant Support Branch Division of Reactor Safety

Report Details

3. SAFEGUARDS
Cornerstone: Physical Protection (PP)

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspector performed an in-office review of the following Physical Security Plan changes to determine if they decreased the effectiveness of the Physical Security Plan and to determine if the requirements of 10 CFR 50.54(p) were met:

- Physical Security Plan, Revision 53, dated September 12, 2001, documented organizational changes.

- Physical Security Plan, Revision 54, dated October 30, 2002, documented that the licensee had welded shut an interior door.

- Independent Spent Fuel Storage Installation Security Plan, Revision 2, dated October 30, 2002, corrected terminology to conform to that used in License Amendment 2 and revised several requirements (i.e., rotation of locks and keys, visual searches of personnel and packages, and illumination) to comply with those listed in the final rulemaking related to 10 CFR 73.51 and the Trojan Safety Evaluation Report.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

The inspector presented the inspection results to Mr. E. Ford, Licensing Engineer, during a telephonic exit conference call on April 23, 2003. The licensee acknowledged the information presented. No proprietary information was identified.