UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

February 1, 2007

Enclosure 2 contains Safeguards Information. When Separated from Enclosure 2, this document and Enclosure 1 are not Safeguards Information.

Mr. William Levis
President and Chief Nuclear Officer
PSEG LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000272/2006005 AND 05000311/2006005

Dear Mr. Levis:

On December 31, 2006, the US Nuclear Regulatory Commission (NRC) completed an inspection at the Salem Nuclear Generating Station. The enclosed integrated inspection report documents the inspection results, which were discussed on January 5, 2007, with you and other members of your staff.

The inspection examined 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The report documents two NRC-identified findings and one self-revealing finding of very low safety significance (Green). This report also documents, in Enclosure 2, one finding of very low security significance (Green) as determined by the Physical Protection Significance Determination Process. The deficiency was promptly corrected or compensated for, and the plant was in compliance with applicable physical protection and security requirements within the scope of this inspection before the inspectors left the site. These findings were determined to involve violations of NRC requirements. Additionally, licensee-identified violations which were determined to be of very low safety significance are included in the report. However, because of the very low safety significance and because they are entered into your corrective action

WARNING: Violation of Section 147 of the Atomic Agency Act, "Safeguards Information," is subject to civil and criminal penalties.	Safeguards information determination made by: Name/Title James M. Trapp, Chief Organization RI/DRS/PSB1 Basis DG-SG-1: (300) Physical Protection Program Signature **/RA/** Date 02/01/07

program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Salem Nuclear Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mel Gray, Chief
Projects Branch 3
Division of Reactor Projects

Docket Nos: 50-272; 50-311
License Nos: DPR-70; DPR-75

Enclosure: 1. Inspection Report 05000272/2006005 and 05000311/2006005 w/Attachment: Supplemental Information
2. Inspection Report 05000272/2006005, 05000311/2006005 and 05000354/2006005 w/Attachment: Supplemental Information **(Contains Safeguards information (SGI))**

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	50-272, 50-311
License Nos:	DPR-70, DPR-75
Report No:	05000272/2006005 and 05000311/2006005
Licensee:	Public Service Enterprise Group Nuclear LLC
Facility:	Salem Generating Station, Units 1 & 2
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	October 1, 2006 through December 31, 2006
Inspectors:	D. Schroeder, Senior Resident Inspector H. Balian, Resident Inspector B. Welling, Senior Project Engineer S. McCarver, Project Engineer D. Tifft, Project Engineer T. O'Hara, Reactor Inspector J. Kulp, Reactor Inspector J. Schoppy, Jr., Senior Reactor Inspector J. Furia, Senior Health Physicist G. Johnson, Operations Engineer
Approved By:	Melvin K. Gray, Chief Projects Branch 3 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000272/2006005, 05000311/2006005; 10/01/2006 - 12/31/2006; Salem Generating Station Units 1 and 2; Equipment Alignment, Inservice Inspection, Refueling and Outage.

The report covered a 13-week period of inspection by resident inspectors, and announced inspections by regional specialists and regional projects inspectors. Three Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," in that corrective actions established in 1998 to identify, clean, and inspect Unit 2 reactor coolant system (RCS) instrument tubing were not implemented. Because these corrective actions were not implemented, three through-wall cracks were identified in RCS instrument tubing in October 2006.

  This finding is more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shut down as well as power operations. The inspectors determined that the finding was of very low safety significance (Green) using a Phase 1 screening in Appendix A of Inspection Manual Chapter 0609, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." Assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function. The finding has a cross-cutting aspect in area of problem identification and resolution, because PSEG did not take appropriate corrective actions, in 1998 and 2005, to address these safety issues in a timely manner, commensurate with their safety significance and complexity. (Section 1R08)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because PSEG did not adequately implement procedural controls for scaffold construction in safety-related areas. This performance deficiency had the potential to adversely impact the upper bearing cooling supply to five of the six Unit 2 service water (SW) pumps and three of the six Unit 1 SW pumps. Once identified, PSEG corrected the scaffold deficiencies.

The issue screened as more than minor based on NRC Inspection Manual Chapter (IMC) 0612, Appendix E, "Examples of Minor Issues and Cross-Cutting Aspects," Example 4.a, because the inspectors identified multiple examples where there was not an engineering seismic impact evaluation to demonstrate no adverse effect on safety-related SW equipment. The finding was determined to be of very low safety significance (Green) because the performance deficiency was not a design deficiency or qualification deficiency; did not represent an actual loss of safety function of a system; did not represent an actual loss of safety function of a single train for greater than the Technical Specification allowed outage time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment; and did not screen as potentially risk significant due to seismic, flooding or a severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance because PSEG personnel did not follow procedures. (Section 1R04)

Cornerstone: Barrier Integrity

- Green. A self-revealing non-cited violation of Salem Technical Specification 6.8.1.b, "Procedures and Programs" was identified when PSEG discovered that an irradiated fuel assembly was incorrectly positioned into the spent fuel pool (SFP) and subsequently transferred without authorization during the reactor core offload of Salem Unit 2's fifteenth refueling outage. Contrary to procedural requirements, PSEG did not ensure that the SFP crane operator used a working copy of the applicable transfer sheets, fuel handling technicians did not properly document a fuel movement irregularity and then transferred a fuel assembly within the SFP without fully apprising the fuel handling senior reactor operator (SRO) or reactor engineer (RE) of the circumstances and, finally, PSEG did not ensure that spent fuel manipulations in the SFP were supervised by a qualified SRO or RE.

  This finding is more than minor because it affected the configuration control attribute of the barrier integrity cornerstone. Specifically, mispositioned fuel in the SFP increases the likelihood of an unanalyzed condition in the SFP and a potential impact on the fuel cladding barrier. An increased likelihood of an unanalyzed condition existed because SFP activities were conducted such that more than one fuel assembly could have been incorrectly positioned. This finding was evaluated by the significance determination process of Inspection Manual Chapter (IMC) 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria" because neither IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations"; nor IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," apply to the spent fuel pool. NRC management determined the finding was of very low safety significance because the deficiency did not cause actual degradation of plant systems, structures or components. Specifically, PSEG analysis demonstrated that the incorrectly positioned fuel assembly was in an acceptably safe location for each move. This finding has a cross-cutting aspect in the area of human performance because PSEG did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. (Section 1R20)

B. Licensee Identified Violations

Violations of very low safety significance, which were identified by PSEG have been reviewed by the inspectors. Corrective actions taken or planned by PSEG have been entered into PSEG's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period at 100 percent (%) power. Operators reduced reactor power to 94% on October 18, 2006, at the request of the transmission system operator (TSO). Operators returned Unit 1 to 100% power on October 20, 2006. Operators reduced reactor power to 82% on November 17, 2006, in response to an emergent trip of the 13A circulating water pump in accordance with plant procedures. Operators returned Unit 1 to 100% power on November 19, 2006.

Unit 2 began the period at 88% power, in end-of-cycle coast-down preceding the fifteenth refueling outage of Unit 2 (S2R15). Operators completed a reactor shutdown on October 10, 2006, to begin S2R15. Operators returned Unit 2 to 100% power on November 5, 2006. Unit 2 remained at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope (1 sample)

The inspectors reviewed the scope of PSEG's cold weather preparations to verify they adequately prepared equipment to operate reliably in freezing conditions. Specifically, the inspectors interviewed engineering and operations personnel, and walked down portions of the service water intake structure, temporary air compressors, and the component cooling water system. The inspectors verified that heat tracing and insulation used to protect these systems were functional and that system conditions were adequate to support operation in cold weather. The documents reviewed during this inspection are listed in the attachment. This inspection satisfied one inspection sample for the onset of cold weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown (3 samples)

The inspectors performed partial walkdowns of four systems comprising three samples to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors focused their review on potential discrepancies that could impact the function of the system, and therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down control systems components, and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that PSEG personnel had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program. Documents reviewed are listed in the attachment. The following systems were walked down:

- Unit 2 service water (SW) header No. 21 during the No. 22 SW header outage;
- Unit 1 and Unit 2 spent fuel pool (SFP) cooling systems concurrent with Unit 2 core offload during refueling outage S2R15; and
- Unit 2 SW supply and return to 2A and 2C emergency diesel generators (EDG) during replacement of 22SW35 (SW supply to 2B EDG).

.2 Complete Walkdown (71111.04S - 1 sample)

a. Inspection Scope

The inspectors conducted one complete walkdown of accessible portions of the Unit 2 service water (SW) system on October 30 through November 3, 2006. The walkdown included SW piping in containment, safety-related SW pipe tunnels, SW valve rooms (78' inner penetration area), residual heat removal pump rooms, and control room instrumentation panels. The inspectors used PSEG procedures and other documents to verify proper system alignment and functional capability. Documents reviewed are listed in the attachment.

The inspectors also verified SW electrical power requirements, labeling, operator workarounds, hangers and support installation, and associated support systems status. The walkdowns also included evaluation of system piping and equipment against the following considerations:

- Oil reservoir levels in normal band;
- Snubber hydraulic fluid leakage;
- Hanger functionality;
- Long-term scaffold construction and placement; and
- Valve alignment and integrity.

In addition, the inspectors reviewed outstanding maintenance work orders to verify that the deficiencies did not significantly affect the SW system function and were being identified and appropriately resolved.

b. Findings

Introduction: The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because PSEG did not properly implement procedural controls for scaffold construction in safety-related areas.

Description: On November 2, 2006, the inspectors identified that scaffolding in Unit 2 SW bay number 2 was in contact with safety-related SW piping. Specifically, the scaffold poles and decking surrounded and contacted 1-½ inch SW alternate cooling piping. The inspectors identified a similar condition in Unit 2 SW bay No. 4, limited to a single point of contact.

Contact between safety-related piping and scaffolding raises the potential for adverse effects on the piping during a seismic event and could cause fretting wear due to vibrations during SW pump operation. Both the number 2 and 4 service water bay alternate cooling piping branches off the SW traveling water screen (TWS) spray wash supply line and is upstream of the common upper motor bearing cooling supply line for the No. 21, 22, 23, 25, and 26 SW pump motors. Upper bearing cooling is an essential support system for these SW pump motors. The 24 SW pump motor is air-cooled and not susceptible to a loss of upper bearing cooling supply. In addition, a break in the alternate cooling pipe may divert SW TWS spray wash flow essential to SW pump operability.

Based on this concern, the inspectors reviewed procedure SH.MD-DG.ZZ-0023, "Scaffold Erection, Modification and Dismantling Desk Top Guide." SH.MD-DG.ZZ-0023 requires a minimum of one inch clearance between scaffolding components and safety-related components except for piping greater than or equal to two inches in diameter (Attachment 4, Section C.2). On November 2, 2006, the inspectors discussed this issue with PSEG. PSEG initiated corrective action notification 20303540. PSEG determined that the scaffolds were not in compliance with SH.MD-DG.ZZ-0023 requirements and made necessary corrections. The inspectors independently verified the corrections on November 3, 2006.

On November 13, 2006, the inspectors identified that a scaffold assembly erected in the Unit 1 SW bay number 1 was in contact with 1-½ inch safety-related SW piping at two locations. Both locations were piping that supplied upper motor bearing cooling water to the 11, 12, and 13 SW Pumps. On November 15, 2006, PSEG reported to the inspectors that these discrepancies were corrected. However, on November 16, 2006, the inspectors found that only one of the contact points had been corrected. PSEG subsequently corrected the second contact point.

PSEG subsequently evaluated the conditions for past operability. PSEG determined by qualitative analysis that the contact between safety-related SW piping and scaffolding did not affect operability. PSEG's evaluation considered the rigidity of the scaffold assemblies and strength of the SW piping.

Analysis: The inspectors determined that PSEG's failure to adequately implement scaffold construction procedure requirements was a performance deficiency that was reasonably within PSEG's ability to foresee and correct. Based on a review of IMC 0612, Appendix E, "Examples of Minor Issues and Cross-Cutting Aspects," Example 4.a, the inspectors noted that the issue screened as more than minor because the inspectors identified multiple examples where there was not an engineering seismic impact evaluation to demonstrate no adverse effect on safety-related SW equipment. In accordance with NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency; did not represent an actual loss of safety function of a system; did not represent an actual loss of safety function of a single train for greater than the Technical Specification allowed outage time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment; and did not screen as potentially risk significant due to seismic, flooding or a severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance because PSEG personnel did not follow procedures.

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, in November 2006, PSEG failed to adequately implement procedure SH.MD-DG.ZZ-0023 requirements for scaffold construction in safety-related areas at Salem Units 1 and 2. Specifically, scaffold assemblies in SW bay numbers 1, 2, and 4 contacted safety-related 1-½ inch piping. This did not comply with the minimum clearance of one inch required by SH.MD-DG.ZZ-0023. Because this finding is of very low safety significance and has been entered into the corrective action program as notification 20303540, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000272&311/2006005-01, Inadequate Procedure Implementation for Scaffold Construction)

1R05 Fire Protection (71111.05)

Fire Protection - Tours

a. Inspection Scope (10 samples)

The inspectors conducted a tour of the ten areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustible material and ignition sources were controlled in accordance with PSEG's administrative procedures; fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with PSEG's fire plan. Documents reviewed are listed in the attachment.

- Unit 1 and Unit 2 Pre-Fire Plan FRS-II-711, Fuel Handling Buildings;
- Unit 1 and Unit 2 Pre-Fire Plan FRS-II-721, Fuel Handling Buildings;
- Unit 1 and Unit 2 Pre-Fire Plan FRS-II-521, Inner Penetrations;
- Unit 1 and Unit 2 Pre-Fire Plan FRS-II-441, Relay Rooms;
- Unit 2 Pre-Fire Plan FRS-II-411, Auxiliary building; and
- Unit 2 Pre-Fire Plan FRS-II-611, Containment during RFO S2R15.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Internal Flooding

a. Inspection Scope (1 sample)

Internal Flooding Review

The inspectors evaluated internal flood protection measures for below grade portions of both auxiliary buildings. The areas were walked down to assess operational readiness of various features to protect vital systems from internal flooding. These features included plant drains, flood barrier curbs, and wall penetration seals. The inspectors also reviewed the results of the most recent flood barrier penetration seal inspection and notifications associated with flood protection measures. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

Resident Inspector Annual Review (1 sample)

The inspectors reviewed performance data and interviewed the PSEG program manager responsible for implementation of the NRC Generic Letter 89-13 program to verify that potential heat exchanger or heat sink deficiencies were identified and that PSEG adequately resolved heat sink performance problems. Specifically, the inspectors reviewed the 22 component cooling (CC) heat exchanger performance data. Inspectors evaluated trending data and verified equipment would perform satisfactorily under design basis conditions. The method of performance monitoring was compared against NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” and EPRI NP-7552, “Heat Exchanger Performance Monitoring Guidelines,” for conformance to these guidance documents. Additional documents reviewed are listed in the attachment.

The inspectors walked down the selected components and the service water intake structure to assess the general material condition of the selected heat exchangers and associated service water components. The inspectors also inspected the internal components of the 22 component cooling heat exchanger, which was open for preventive maintenance, and observed the type and quantity of material present within the heat exchanger. The inspectors reviewed photographs of the 22 CC heat exchanger internals taken before and after cleaning and preservation activities. The inspectors reviewed a sample of notifications related to service water heat exchangers to ensure that problems related to these components were appropriately identified, characterized, and corrected. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope (8 samples)

The inspectors observed selected samples of nondestructive examination (NDE) activities in process. Also, the inspectors reviewed additional samples of completed NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage. The observations and documentation reviews were performed to verify the activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspectors reviewed a sample of inspection reports and condition reports (notifications) initiated as a result of problems identified during inservice inspection (ISI) examinations. Also, the inspectors evaluated the effectiveness of the resolution of problems identified during selected ISI activities. Documents reviewed are listed in the attachment.

The inspectors reviewed PSEG's boric acid corrosion control program. Additionally, the inspectors observed PSEG's boric acid walkdown inspection in containment. The walkdown inspections were thorough, well organized, and indications of boric acid leakage were evaluated in accordance with PSEG's program for documentation in the corrective action (notification) process. Additionally, the inspectors reviewed a sampling of notifications for correct evaluation and/or further engineering analysis and/or resolution. (1 sample)

The inspectors observed the performance of two in-process NDE activities and reviewed documentation and examination reports for an additional fourteen NDE activities. The inspectors reviewed 2 samples of welding activities on a pressure boundary component and, reviewed the package for a repair performed in accordance with the ASME Code during the previous operating cycle. These observations and reviews covered ultrasonic testing (UT), visual examination (VT), penetrant testing (PT) and magnetic particle testing (MT) techniques. (4 samples)

The inspectors reviewed inspection data sheets and documentation for manual UT activities to verify the effectiveness of the examiner, process, and equipment to identify degradation of risk significant systems, structures and components and to evaluate the activities for compliance with the requirements of ASME Code, Section XI. The inspectors reviewed a sample of work orders for repairs made to plant components to evaluate compliance with the requirements of the ASME Code, Section XI. (1 sample)

The inspectors reviewed four samples of NDE evaluations which had been initially rejected and subsequently accepted after evaluation. The inspectors also reviewed the radiographs and the examiners' interpretation of indications on five main steam system component welds. (1 sample)

The inspectors reviewed report "Steam Generator Degradation Report for Salem Unit 2," dated September 2006. This report documented the steam generator degradation, measured in refueling outage 2SR14, and gave the technical basis for the inspections conducted during this refueling outage. The inspectors reviewed the data collection and interpretation activities to verify compliance with procedures. The inspectors reviewed the results of the eddy current examinations and all notifications generated as a result of the inspections. The inspectors also participated in a conference call on October 18, 2006, with PSEG and NRC personnel from the Office of Nuclear Reactor Regulation, discussing the results obtained and the status of the eddy current inspections up to that time. (1 sample)

The inspectors verified through review of PSEG records and PSEG correspondence with the NRC that bare metal visual inspection of the reactor vessel lower head penetrations per Temporary Instruction (TI) 2515/152 was not required during this outage.

b. Findings

Introduction: The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," in that corrective actions established in 1998 to identify, clean, and inspect Unit 2 reactor coolant system (RCS) instrument tubing were not implemented. Because these corrective actions were not implemented, three through-wall cracks were identified in RCS instrument tubing in October 2006.

Description: In July 1998, PSEG discovered nine leaks on RCS instrumentation tubing in the Unit 2 containment. Eight leaks were discovered in six separate RCS instrument tubing sections and one leak was discovered in the RCS sampling system tubing. PSEG determined that the cause was initiated as cracking on the outside of the tubing which progressed through wall to the inside surface by a transgranular stress corrosion cracking mechanism. This condition was reported via Licensee Event Report (LER) 05000311/1998007-00, on August 27, 1998. The root cause analysis further attributed the cracking to the presence of local residual stresses in the presence of contaminants, such as halogens, phosphate, and sulfate on the outside surface of the tubing. The analysis identified the source of the contaminants to be service water. Several corrective actions were specified to correct the causes and to assess the effectiveness of these actions. All corrective actions were planned for completion during the subsequent refueling outage in 1999. The inspectors observed that no documentation exists that the specified corrective actions were completed in 1999 or that the actions were continued after 1999. PSEG did not identify corrective actions to control the intrusion of contaminants through the service water system.

In April 2005, four through-wall leaks of instrument tubing were identified during the refueling outage (notifications 20231322, 20233095, 20233096, and 20236992). PSEG determined that these leaks were caused by the same mechanism which caused the tubing leaks in 1998. PSEG also determined that service water contaminants existed on the tubing in 2005 and that they were a significant contributing factor. In October 2006 three new RCS instrument tubing leaks were identified. The stainless steel tubing was replaced and sent to the lab for analysis. PSEG concluded the structural integrity of the tubing had not been compromised, and the cause of the leaks was determined to be transgranular stress corrosion cracking, the same mechanism found in earlier failures.

Analysis: A performance deficiency was identified in that PSEG did not implement corrective actions created in 1998 and 2005 as a result of evaluating similar instrument tubing failures. These actions included revising maintenance procedures to control the the intrusion of contaminants. Consequently, PSEG identified three through-wall leaks on RCS instrument tubes in October 2006. This finding is more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shut down as well as power operations. The inspectors determined that the finding is of very low safety significance (Green) using a Phase 1 screening in Appendix A of Inspection Manual Chapter 0609, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” Assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function. The finding has a cross-cutting aspect in the area of problem identification and resolution; specifically, PSEG did not implement corrective actions created in both 1998 and 2005, to address these safety issues in a timely manner, commensurate with their safety significance and complexity.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and nonconformances are promptly identified and corrected. Contrary to the above, PSEG did not fully implement corrective actions for leaks in RCS instrument tubing identified in July 1998, which resulted in three through-wall leaks in October 2006. Because the finding is of very low significance and has been entered into PSEG’s corrective action program (notification 20308078), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2006005-02, Failure to Implement Effective Corrective Actions for Reactor Coolant System Tubing Leaks)**

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope (1 sample)

The inspectors observed a simulator training scenario conducted on November 28, 2006, to assess operator performance and training effectiveness. The scenario included a volume control tank level failure, a heater drain pump trip, de-energized 2A vital bus, 23 reactor coolant pump number 1 seal failure, and a steam leak inside containment. The inspectors verified operator actions were consistent with operating, alarm response, abnormal and emergency procedures. The inspectors assessed simulator fidelity and verified that evaluators identified deficient operator performance where appropriate. The inspectors observed the simulator instructor's critique of operator performance. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (3 samples)

The inspectors reviewed performance monitoring and maintenance effectiveness issues for one component and two systems. The inspectors assessed whether PSEG was adequately monitoring equipment performance to ensure that preventive maintenance was effective. The inspectors verified that the components were monitored in accordance with the maintenance rule program requirements. The inspectors compared documented functional failure determinations and unavailability hours to those being tracked by PSEG to evaluate the effectiveness of PSEG's condition monitoring activities and to determine whether performance goals were being met. The inspectors reviewed applicable work orders, corrective action notifications, and preventive maintenance tasks. Documents reviewed are listed in the attachment. The following three samples were completed:

- Unit 1 control area chiller;
- Unit 2 electro-hydraulic control (EHC) system; and
- Unit 1 and Unit 2 auxiliary building ventilation (ABV) systems.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope (7 samples)

The inspectors reviewed seven maintenance activities to verify that the appropriate risk assessments were performed as required by 10 CFR 50.65(a)(4) prior to removing equipment for work. The inspectors reviewed the applicable risk evaluations, work schedules and control room logs for these configurations to verify that concurrent planned, and emergent maintenance and test activities did not adversely affect the plant risk already incurred with these configurations. PSEG's risk management actions were reviewed during shift turnover meetings, control room tours, and plant walkdowns. The inspectors also used PSEG's on-line risk monitor (Equipment Out-Of-Service Workstation) and safety functional assessment trees (SFATs) to evaluate the risk associated with the plant configuration and to assess PSEG's risk management. In addition, the inspectors reviewed notifications documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed are listed in the attachment. The following plant configurations were assessed:

- 1C emergency diesel generator (EDG) and No. 22 SW header maintenance;
- 11 and 15 service water pumps out of service concurrently;
- 2A EDG and 13 station power transformer (SPT) out of service concurrently;
- Adequacy of shutdown cooling, inventory control and electric power key safety functions during the S2R15 mid-loop condition before core offload;
- Adequacy of the reactivity control key safety function during S2R15 core reload;
- Adequacy and completeness of equipment protection during the S2R15 mid-loop condition after S2R15 core reload and reactor coolant system vacuum fill; and
- Operability and risk assessment of the Unit 1 component cooling (CC) water system during planned maintenance that required removing the 12B CC heat exchanger from service.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (6 samples)

The inspectors reviewed six operability determinations for degraded or non-conforming conditions associated with:

- Notification (NOTF) 20284189, 22 service water strainer blowdown valve remained midposition;
- NOTF 20303419, Unit 2 quadrant power tilt ratio;
- Work Orders (WO) 30122829 and 30122727, source range nuclear instrumentation during Unit 2 core alterations;
- NOTF 20305609, automatic start of the 21 emergency air conditioning supply fan caused by malfunctioning air flow pressure transmitter;
- NOTF 20301991, 21 centrifugal charging pump high vibration during full flow testing; and
- NOTF 20303770, Unit 2 high pressure safety injection with backleakage through centrifugal charging pump discharge check valve 22CV47.

The inspectors reviewed the technical adequacy of the operability determinations to verify the conclusions were justified. The inspectors also walked down accessible equipment to corroborate the adequacy of PSEG's operability determinations. Additionally, the inspectors reviewed other PSEG identified safety-related equipment deficiencies during this report period and assessed the adequacy of their operability screenings. Notifications and documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope (1 sample)

The inspectors reviewed permanent modifications to the Unit 2 main steam line radiation monitoring system (2R46) under change number 80057587. This review included system walkdowns, interviews with plant engineers and operators, and functional comparison of the new 2R46 monitors to the original 2R46 monitors. Finally, the review included verification that the new 2R46 monitors satisfied Salem's current licensing basis.

This modification replaced main steam line radiation monitors with four new radiation monitors (2R46A through D). The original 2R46 monitors included an off-line sampling system which drew a steam sample from each main steam line and measured activity in the sample. Consequently, the original monitors required a chilled water cooling system. The new monitors are environmentally qualified shielded ion chambers mounted adjacent to the main steam line to directly measure activity in the process steam. Therefore, the need for a radiation monitor chilled water cooling system is eliminated.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (7 samples)

The inspectors observed portions of and/or reviewed results of seven post-maintenance tests for the following equipment:

- Work orders (WO) 30033947, 60065649; 24 containment fan cooler unit (CFCU) internal inspection and preservation activities during refueling outage (RFO) S2R15;
- WO 30102532; 21 reactor coolant pump (RCP) seal replacement during RFO S2R15;
- WO 60050018, 22 centrifugal charging pump speed changer replacement during S2R15;
- WO 30130351, 21 centrifugal charging pump discharge check valve 2CV47 open and inspect, including vendor recommended repair;
- WO 50002207, Unit 2 containment integrated leak rate test during S2R15;
- WO 50085912, 2C emergency diesel generator (EDG) following eighteen month periodic maintenance; and

- WO 60066694, emergent repair to ASME weld upstream of containment spray valve 2CS61.

The inspectors assessed whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness, consistent with design and licensing basis documentation; (4) test instrumentation had current calibration, range, and accuracy for the application; (5) tests were performed, as written, with applicable prerequisites satisfied; and (6) equipment was returned to an operational status and ready to perform its safety function. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

The inspectors reviewed the schedule and risk assessment documents associated with the Salem Unit 2 refueling outage to confirm that PSEG appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth systems and barriers. Prior to the refueling outage, the inspectors reviewed PSEG's outage risk assessment to identify risk significant equipment configurations and determine whether planned risk management actions were adequate. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored PSEG controls over the outage activities listed below. The inspectors verified that cool down rates were within Technical Specification (TS) limitations.

The inspectors observed the Unit 2 reactor coolant system (RCS) draining to the mid-loop condition on October 14, 2006. Reactor coolant system inventory controls and contingency plans were reviewed by the inspectors to determine whether they met TS requirements and provided for adequate inventory control. The inspectors reviewed procedures and observed portions of activities in the control room when the unit was in reduced inventory modes of operation, including mid-loop operations. The inspectors verified that level and core temperature measurement instrumentation was installed and operational. Calculations that provide time-to-boil information were also reviewed for RCS reduced inventory conditions as well as the spent fuel pool during increased heat load conditions.

The inspectors also observed conditions within containment for indications of unidentified leakage and damaged equipment. The inspectors verified that PSEG managed the outage risk commensurate with the outage plan. The inspectors periodically observed refueling activities from the refueling bridge in containment and the spent fuel pool to verify refueling gates and seals were properly installed and determine whether foreign material exclusion boundaries were established around the reactor cavity. Core offload and reload activities were periodically observed from the control room and refueling bridge to verify whether operators adequately controlled fuel movements in accordance with procedures.

The inspectors verified that tagged equipment was properly controlled and equipment configured to safely support maintenance work. Specifically, the inspectors walked down service water (SW) system tagouts for isolating one SW header and hardening the remaining inservice SW header. The inspectors also walked down a tagout supporting replacement of 22SW35 (2B emergency diesel generator). Equipment work areas were periodically observed to determine whether foreign material exclusion boundaries were adequate. During control room tours, the inspectors verified that operators maintained adequate RCS level and temperature and that indications were within the expected range for the operating mode.

The inspectors determined whether offsite and onsite electrical power sources were maintained in accordance with TS requirements and consistent with the outage risk assessment. Periodic walkdowns of portions of the switchyard, onsite electrical buses and the EDGs were conducted during risk significant electrical configurations. The inspectors verified through routine plant status activities that the decay heat removal safety function was maintained with appropriate redundancy as required by TS and consistent with PSEG's outage risk assessment. During core offload conditions, the inspectors periodically determined whether the fuel pool cooling system was performing in accordance with applicable TS requirements and consistent with PSEG's risk assessment for the refueling outage.

Containment status and procedural controls were reviewed by the inspectors during fuel offload and reload activities to verify that TS requirements and procedure requirements were met for containment. Specifically, the inspectors verified that during fuel movement activities, personnel, materials and equipment were staged to close containment penetrations as specified in the licensing basis.

The inspectors conducted a thorough walk down of containment prior to reactor startup. Areas of containment where work was completed were inspected for evidence of leakage and to ensure debris which could block containment sumps was removed. Containment integrity was verified by observing portions of the containment integrated leak rate test. Portions of mode changes and reactor startup were observed and reviewed for compliance with applicable procedures and Technical Specifications.

The inspectors reviewed applicable documents associated with the Salem Unit 2 refueling outage as listed in the attachment.

b. Findings

Introduction: A self-revealing Green non-cited violation (NCV) of Salem Generating Station Technical Specification 6.8.1.b, "Procedures and Programs," was identified when PSEG discovered that an irradiated fuel assembly was incorrectly positioned into the spent fuel pool (SFP) and subsequently transferred without authorization during the reactor core offload of Salem Unit 2's fifteenth refueling outage.

Description: On October 17, 2006, during refueling operations, fuel handling personnel lowered and unlatched spent fuel assembly NS20 in SFP rack location AA-5 instead of its assigned location of BC-32. Shortly thereafter, the fuel handling personnel latched fuel assembly NS20 and moved it to SFP rack location BC-32. Fuel assembly NS20 was stored in an incorrect SFP rack location for approximately eighteen minutes.

PSEG determined that during the core offload, the SFP crane operator inappropriately relied on a computer driven billboard display to determine where the irradiated fuel assembly was to be stored. The SFP crane operator did not use a working copy of the Item Control Area (ICA) transfer sheets as required by SC.RE-FR.ZZ-0001, "Fuel Handling." A working copy of the ICA transfer sheets was in use by another member of the SFP fuel handling team. However, this member of the team did not stop spent fuel manipulations when a disagreement between the ICA transfer sheets and the computer driven billboard display manifested itself.

PSEG determined that the SFP crane operator moved fuel assembly NS20 to SFP rack location AA-5 because the computer driven billboard display indicated AA-5. The SFP crane operator then lowered and unlatched fuel assembly NS20 in SFP rack location AA-5. Concurrently, other members of the fuel handling team worked to resolve the disagreement between the ICA transfer sheets and the computer driven billboard display and determined the computer driven billboard display was one step ahead of the actual fuel transfer sequence. The error was communicated to the SFP crane operator after fuel assembly NS20 was unlatched in SFP rack location AA-5. Subsequently, the SFP crane operator latched fuel assembly NS20 and moved it to SFP rack location, BC-32. The inspectors observed this movement was completed without the procedurally required authorization of an ICA transfer sheet or a Fuel Assembly Handling Deviation Report approved by a qualified senior reactor operator (SRO) or a qualified reactor engineer (RE).

PSEG entered this event into the corrective action program as notification 20300992 and conducted a quick human performance investigation (QHPI). In response to the findings of the QHPI, adjustments were made to SFP crane operating personnel, and retraining was provided to SFP crane operators, fuel handling technicians and specialists. A working copy of the ICA transfer sheets was made available to the SFP crane operator. PSEG further provided the direct oversight of a qualified RE in the SFP area.

The inspectors identified instances whereby PSEG did not adhere to established procedures. First, SC.RE-FR.ZZ-0001, "Fuel Handling," step 2.1.5 required the SFP crane operator to maintain a working copy of the applicable ICA transfer sheets on the bridge crane when moving fuel. PSEG did not require the crane operator to maintain a working copy of the applicable ICA transfer sheets on the bridge crane before this incident. PSEG did require the crane operator to maintain a working copy of the ICA transfer sheets during all subsequent fuel assembly movement in the SFP.

Second, SC.RE-FR.ZZ-0019, "Refueling," step 5.3.2 required that the reactor core be off-loaded per core offload transfer sheets. Similarly, SC.RE-FM.ZZ-0001, "Special Nuclear Material Control and Accounting," step 5.2.4 required that special nuclear material be transferred per authorized ICA transfer sheets. Fuel handling technicians did not transfer fuel assembly NS20 per core offload transfer sheets or ICA transfer sheets. Specifically, fuel assembly NS20 was first placed in SFP rack location AA-5 and then moved to SFP location BC-32.

Third, SC.RE-FM.ZZ-0001, "Special Nuclear Material Control and Accounting," in a note preceding step 5.2.4, required authorization of the refueling SRO or RE to change the transfer sequence. SC.RE-FR.ZZ-0019, step 3.1.19 required documentation of all fuel handling irregularities on a "Form-1, Fuel Assembly Handling Deviation Report." Both the refueling SRO and RE are required to initial the Form-1. In addition, SC.RE-FR.ZZ-0001, step 3.1.5 required fuel handling technicians to stop work if an unexpected response arose. Further, SC.RE-FR.ZZ-0019, step 3.1.14 required the fuel handling technicians to stop work and notify the refueling SRO and reactor engineering if any inconsistency in the loading sequence or recording of fuel movements were identified. Fuel handling technicians did not stop work and fully apprise the refueling SRO or RE of the circumstances regarding offloading of fuel assembly NS20. Consequently, two additional fuel assemblies were offloaded before the refueling SRO became aware of the incorrect placement. Moreover, the transfer of fuel assembly NS20 from AA-5 to BC-32 was not properly authorized.

Additionally, S2.OP-IO.ZZ-0010, "Spent Fuel Manipulations," step 2.2 required the supervision of a qualified SRO or RE when manipulating spent fuel in the SFP. Following this incident, the RE was stationed at the SFP for all manipulations of spent fuel in the SFP.

Analysis: The inspectors determined that the incorrect placement and subsequent unauthorized transfer of fuel assembly NS20 followed by two more fuel assembly transfers was a performance deficiency. The finding is more than minor because it affected the configuration control attribute of the barrier integrity cornerstone. Specifically, incorrect placement of fuel in the SFP increases the likelihood of an unanalyzed condition in the SFP and a potential impact on the fuel cladding barrier. An increased likelihood of an unanalyzed condition existed because SFP activities were conducted such that more than one fuel assembly could have been incorrectly positioned. This finding was evaluated by the significance determination process of Inspection Manual Chapter (IMC) 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria" because neither IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations"; nor IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," apply to the spent fuel pool. NRC management determined the finding was of very low safety significance because the deficiency did not cause actual degradation of plant systems, structures or components. Specifically, PSEG analysis demonstrated that placing fuel assembly NS20 into SFP location AA-5 was an acceptably safe location. This finding has a cross-cutting aspect in the area of human performance because PSEG did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Specifically, the inspectors concluded that inadequate oversight resulted in an additional, unauthorized fuel assembly move after the initial error that resulted in the issue being more than minor.

Enforcement: Salem Generating Station Technical Specification 6.8.1.b, “Procedures and Programs,” requires, in part, that written procedures be implemented for refueling operations. Specifically, SC.RE-FR.ZZ-0001, “Fuel Handling,” requires the SFP crane operator to use a working copy of the applicable ICA transfer sheets when moving fuel. SC.RE-FR.ZZ-0019, “Refueling,” and SC.RE-FM.ZZ-0001, “Special Nuclear Material Control and Accounting,” requires the reactor core be off-loaded per ICA transfer sheets. SC.RE-FM.ZZ-0001, “Special Nuclear Material Control and Accounting,” requires authorization of the refueling SRO or RE to change the transfer sequence. Finally, S2.OP-IO.ZZ-0010, “Spent Fuel Manipulations,” requires the supervision of a qualified SRO or RE when manipulating spent fuel in the SFP.

Contrary to the above, on October 17, 2006, PSEG did not properly implement refueling operations procedures. Specifically, PSEG did not ensure that the SFP crane operator used a working copy of the applicable ICA transfer sheets; fuel handling personnel did not document a fuel movement irregularity on a Fuel Assembly Handling Deviation Report, but did transfer fuel assembly NS20 within the SFP without fully apprising the fuel handling SRO or RE of the circumstances to obtain authorization; and, finally, PSEG did not ensure that spent fuel manipulations in the SFP were supervised by a qualified SRO or RE. Because this finding is of very low safety significance and has been entered into the corrective action program as notification 20300992, this violation is being treated as an NCV, consistent with section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2006005-03, Incorrectly Positioned Fuel Assembly)**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (4 samples)

The inspectors observed portions of and/or reviewed results for four surveillance tests to verify, as appropriate, whether the applicable system requirements for operability were adequately incorporated into the procedures and that test acceptance criteria were consistent with procedure requirements, the Technical Specification requirements, the Updated Final Safety Analysis Report (UFSAR), and ASME Section XI for pump and valve testing. Documents reviewed are listed in the attachment. The following surveillance tests were inspected:

- WO 50086489, Safeguard Equipment Cabinet (SEC) Mode Ops Testing 2A Vital Bus;
- WO 50097618, 11 safety injection pump inservice testing;
- S1.OP-ST.RC-0008, reactor coolant system water inventory balance; and
- WO 50099614, 15 service water pump inservice testing.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

The inspectors reviewed temporary modification 70062259, “spent fuel pool high temperature alarm setpoint revision,” installed on October 17, 2006. The inspectors assessed whether PSEG followed its administrative process for implementing the modification, NC.DE-AP.ZZ-0030, “Control of Temporary Modifications,” and verified that the temporary modification did not adversely impact the operation and performance of the associated system. The inspectors verified that the modification did not affect the operators’ response to abnormal or emergency conditions.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (8 samples)

The inspectors reviewed radiation work permits (RWPs) used to access high radiation areas and identified what work control instructions or control barriers have been specified. The inspectors reviewed electronic personal dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy.

The inspectors reviewed RWPs for airborne radioactivity areas with the potential for individual worker internal exposures of >50 mrem Committed Effective Dose Equivalent [CEDE] (20 DAC-hrs). The inspectors verified barrier integrity and engineering controls performance (e.g., HEPA ventilation system operation).

Based on PSEG’s schedule of work activities during the Unit 2 refueling outage (S2R15), the inspectors selected two jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation. The work selected was nozzle dam/bowl work/channel head entries and S2R15 scaffolding activities in the Unit 2 containment. The inspectors reviewed radiological job requirements (RWP requirements and work procedure requirements) and observed job performance with respect to these requirements. The inspectors determined that radiological conditions in the work area were adequately communicated to workers through briefings and postings.

During job performance observations, the inspectors verified the adequacy of radiological controls, such as: required surveys (including system breach radiation, contamination, and airborne surveys), radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls.

For high radiation work areas with significant dose rate gradients (factor of 5 or more), the inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel. The inspectors verified that PSEG controls were adequate.

During job performance observations, the inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors determined that workers were aware of the significant radiological conditions in their workplace, and the RWP controls/limits in place, and that their performance took into consideration the level of radiological hazards present.

During job performance observations, the inspectors observed radiation protection technician performance with respect to radiation protection work requirements. The inspectors determined that they were aware of the radiological conditions in their workplace and the RWP controls/limits, and that their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (2 samples)

Based on scheduled work activities and associated exposure estimates, the inspectors selected two work activities in radiation areas, airborne radioactivity areas, or high radiation areas for observation (Section 2OS1 above). The inspectors evaluated PSEG's use of as low as is reasonably achievable (ALARA) controls for these work activities by performing the following: evaluated PSEG's use of engineering controls to achieve dose reductions, procedures and controls consistent with PSEG's ALARA reviews; verified that sufficient shielding of radiation sources provided for; and verified that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

a. Inspection Scope (1 sample)

The inspectors identified the types of portable radiation detection instrumentation used for job coverage of high radiation area work, other temporary area radiation monitors currently used in the plant, and continuous air monitors associated with jobs with the potential for workers to receive 50 mrem CEDE.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

Cornerstone: Initiating Events (6 samples)

- Unplanned Scrams per 7,000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

For Salem Units 1 and 2, the inspectors reviewed PSEG power history charts, licensee event reports, NRC Monthly Operating Reports, and control room logs to determine whether PSEG had adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred during the previous nine quarters, fourth quarter 2004 to fourth quarter 2006. This number was compared to the number reported for the PI during the current quarter. The inspectors also verified the reported critical hours accuracy. The inspectors interviewed PSEG personnel associated with PI data collection, evaluation, and distribution.

Cornerstone: Occupational Radiation Safety (1 sample)

The inspectors reviewed all PSEG performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed a listing of PSEG action reports for the period January 1, 2006, through October 16, 2006, for issues related to the occupational radiation safety performance indicator, which measures nonconformances with high radiation areas greater than 1R/hr and unplanned personnel exposures greater than 100 mrem total effective dose equivalent (TEDE), 5 rem skin dose equivalent (SDE), 1.5 rem lens dose equivalent (LDE), or 100 mrem to the unborn child.

The inspectors determined if any of these PI events involved dose rates >25 R/hr at 30 centimeters or >500 R/hr at 1 meter. If so, the inspectors determined what barriers had failed and if there were any barriers left to prevent personnel access. For unintended exposures >100 mrem TEDE (or >5 rem SDE or >1.5 rem LDE), the inspectors determined if there were any overexposures or substantial potential for overexposure.

Cornerstone: Public Radiation Safety (1 sample)

The inspectors reviewed a listing of PSEG action reports for the period January 1, 2006, through October 16, 2006, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/qtr whole body or 5 mrem/qtr organ dose for liquid effluents; or 5 mrads/qtr gamma air dose, 10 mrads/qtr beta air dose; or 7.5 mrems/qtr organ doses from I-131, I-133, H-3 and particulates for gaseous effluents.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the Corrective Action Program

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into PSEG's corrective action program. This was accomplished by reviewing the description of each new notification and attending daily management review committee meetings. Documents reviewed are listed in the attachment.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of PSEG's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1. The review also included issues documented in system health reports, corrective maintenance WOs, component status reports, site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the six-month period of May 1, 2006 to December 1, 2006, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results contained in PSEG's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in PSEG's trend report were reviewed for adequacy. The inspectors also evaluated the trend report specified in SPP-3.1, Corrective Action Program. Specific documents reviewed are listed in the attachment.

b. Assessment and Observations

No findings of significance were identified.

The inspectors noted a trend of low level issues entered into the CAP related to the areas of plant status control, valve problems, and fire door deficiencies. The inspectors determined PSEG is aware of these areas identified through this trend review and is appropriately addressing these issues.

.3 Annual Sample: Corrective Actions Related to the Auxiliary Feedwater System (AFW)

a. Inspection Scope

The inspectors reviewed PSEG's progress on corrective actions to address operator actions for the auxiliary feedwater system during station blackout (SBO) or Appendix R fire scenarios. The inspectors confirmed PSEG had issued a purchase order to have a contractor prepare analyses for PSEG to evaluate plant conditions following Station Blackout (SBO) and Appendix R fire scenarios. The resulting analyses will provide PSEG information to address the applicable corrective action program notifications (20277247, 20271973 and 20277379).

The inspectors also reviewed PSEG's corrective actions for notification 20272356, 11 AFW pump low suction trip, which included an action to establish a new calculation for an AFW pump trip setpoint that accounts for the effects of vortexing in the pump suction lines.

b. Findings and Observations

No findings of significance were identified.

PSEG completed a calculation for an auxiliary feedwater pump trip set point which accounts for vortexing on the suction lines. The inspectors reviewed the calculation, SC-AF002-02, Salem Unit 1 & 2 AFW Pumps Suction Pressure Low Setpoint Tornado Armed Trip, Rev. 0, and noted the new setpoint provided reasonable assurance that air entrainment in the pump suction lines would not occur. The inspectors determined that the corrective actions were appropriate and timely.

.4 Annual Sample: Recommendations from Safety Conscious Work Environment Peer Assessment Report

a. Inspection Scope

The inspectors reviewed PSEG's actions to address twenty-two recommendations from the report of the independent peer assessment of the safety conscious work environment (SCWE), which was submitted to the NRC by letter dated May 4, 2006. These recommendations were intended to help PSEG sustain the improvements in the SCWE that were observed by the peer assessment team. In June 2006 the NRC conducted a special inspection of the SCWE, documented in NRC inspection report 05000272;311/2006012 and 05000354/2006011, and concluded that improvements to the work environment at Salem and Hope Creek were substantial and sustainable.

The NRC's Mid-Cycle Performance Review letter for Salem, dated August 31, 2006, stated that the NRC intended to continue to monitor performance in the SCWE during baseline inspections. Further, this letter stated that the NRC would review recommendations from the independent peer assessment, as tracked in the corrective action program, through a baseline problem identification and resolution sample inspection. During this review, the inspectors reviewed corrective action program notifications, supporting documentation, and discussed the actions with station personnel.

b. Findings and Observations

No findings of significance were identified.

The inspectors determined that PSEG had taken appropriate actions to address the twenty-two recommendations from the peer assessment team. All of the recommendations were entered in corrective action program notifications, and prompt actions were taken to resolve the items, with one exception.

The inspectors noted that PSEG did not complete actions to address one of the recommendations. Specifically, the inspectors identified that PSEG had not revised the SCWE section in performance evaluations forms to more specifically align it to behaviors that encourage personnel to raise concerns without fear of retaliation, as recommended by the peer assessment team (Recommendation 1.4.3.1). Instead, the SCWE section of the form prompts managers to evaluate whether the individual "creates conditions and practices which place safety as the highest priority." This statement is not consistent with the recommendation and may lead to confusion between SCWE attributes and industrial safety behaviors. In response to this NRC observation, PSEG placed the item in the corrective action program for resolution (notification 20305890). The inspectors concluded that this issue was minor, because there was no impact on plant equipment, human performance, or the safety conscious work environment.

4OA3 Event Followup (71153 - 4 LER samples)

.1 (Closed) LER 05000311/2006003-00, Manual Reactor Trip due to Elevated Reactor Coolant Pump Seal Leak-Off

This LER describes the reactor trip of September 26, 2006, when operators manually tripped Unit 2 because the 21 reactor coolant pump (RCP) seal return flow rate exceeded six gallons per minute (GPM). PSEG determined that the excessive RCP seal return flow was caused by a chemistry transient induced by RCS chemistry conditions unique to the end of cycle. There were no actual safety consequences associated with this event and all plant systems operated as designed. The inspectors identified no findings and no violations of NRC requirements. PSEG documented the problem in notification 20298370. This LER is closed.

.2 (Closed) LER 05000272/2006002-00, Inoperability of the Auxiliary Feedwater System Due to a Partially Opened Damper

On September 28, 2006, PSEG discovered that auxiliary building ventilation damper 1ABS20 was stuck partially open. PSEG determined that 1ABS20 failed in the partially open position because the associated actuator was damaged by excessive operating air pressure. Corrective actions included correctly setting the 1ABS20 actuator air pressure, inspecting other dampers for proper operation and operating air pressure, and revising maintenance procedures to direct proper setting of the operating air pressure. This finding is more than minor because it had a credible impact on safety, in that if 1ABS20 did not close in response to a steam line break, operability of the auxiliary feedwater system (AFW) would not be assured. The finding affects the Mitigating Systems Cornerstone and is considered to have very low safety significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations because the issue was not a design deficiency or qualification deficiency; did not represent an actual loss of safety function of a system; did not represent an actual loss of safety function of a single train for greater than the Technical Specification allowed outage time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment; and did not screen as potentially risk significant due to seismic, flooding or a severe weather initiating event. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

.3 (Closed) LER 05000311/2006004-00, Fuel Assemblies Misloaded in the Spent Fuel Pool Racks

On October 12, 2006, PSEG discovered that a fuel assembly which did not meet the Technical Specification requirements for unrestricted storage was placed in an unrestricted storage location of the Unit 2 spent fuel pool (SFP). During an extent of condition review, PSEG found another twelve fuel assemblies stored in unrestricted storage locations that did not meet the requirements for unrestricted storage. Twelve of the thirteen misloaded fuel assemblies were moved to acceptable SFP storage locations. The thirteenth fuel assembly was reinserted into the Unit 2 core during S2R15, where it is currently positioned. PSEG plans to prevent recurrence by incorporating appropriate human performance tools and techniques into processes and procedures before next used. This finding is more than minor because it had a credible impact on safety in that mispositioned fuel in the SFP raises the likelihood of an unanalyzed condition in the SFP. The finding affects the Barrier Integrity Cornerstone and is of very low safety significance because PSEG demonstrated that reactivity was acceptably low and within requirements while this condition existed. This licensee identified finding involved a violation of TS 3.7.12, Fuel Assembly Storage in the Spent Fuel Pool. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

.4 (Closed) LER 05000311/2006005-00, Automatic Start of Auxiliary Feedwater Pumps in Mode 4

While in Mode 4, Hot Shutdown, on October 29, 2006, valid reactor protection system (RPS) and engineered safeguards feature (ESF) signals were generated by a low-low water level condition in the 22 steam generator. The low-low water condition in the 22 steam generator occurred because operators tested the 22 main steam isolation valve (22MS167) while the main turbine stop and control valves were open per main turbine testing that was in progress. PSEG subsequently determined this event was caused by insufficient broad oversight and coordination of all control room activities and reviewed all control room activities during the remainder of the outage to identify and resolve operational conflicts. The inspectors determined that this issue is minor because it was an insignificant procedural error with no safety consequence. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

4OA5 Other Activities

.1 Temporary Instruction (TI) 2515/166 - Pressurized Water Reactor Containment Sump Blockage

a. Inspection Scope

The inspectors performed an inspection of modifications to the Unit 2 containment sump in accordance with TI 2515/166. The TI was developed to support the NRC review of licensee activities in response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWR)." Specifically, the inspectors verified implementation of the modifications and procedure changes committed to in the GL response. The inspectors reviewed a sample of the licensing and design documents to verify that they were either updated or in the process of being updated to reflect the modifications. A sample of material specifications, testing and surveillance procedures, operator training lesson plans, and calculations were reviewed to verify that they were updated to reflect the effects of the modification, and the new requirements for the containment sumps and debris generation sources. The inspectors observed construction activities and performed a walkdown of the strainer to verify it was installed in accordance with the approved design change package. Additionally, the inspectors walked down the steam generator blowdown piping where insulation in containment that could be dislodged during a loss of coolant accident was replaced by reflective metal insulation. Finally, the inspectors walked down areas that could pose choke-points that could prevent water from reaching the recirculation sump during a design basis accident.

b. Evaluation of Inspection Requirements:

The TI requested the inspectors to evaluate and answer the following questions:

1. Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 response?

   The inspectors verified that actions implemented by PSEG as described in response to Generic Letter 2004-02 were partially complete as it related to the installation of the sump screen, removal of insulation, evaluation of potential debris sources and the potential for clogging of downstream component due to debris bypass. Additionally, the inspectors found that procedures to programmatically control potential debris generation sources were updated. The inspectors noted that the sump surface area that was installed had a smaller surface area than was discussed in the GL response. PSEG entered this issue into their corrective action program and intends to update the GL response. The inspectors noted that PSEG had not completed the long term downstream effects evaluation or the effects of chemical precipitants on the strainer head loss at the time of the inspection. PSEG identified the bioshield doors on the 78 foot level as potential choke-points that could prevent water from reaching the recirculation sump during a design basis accident. The Post-LOCA Debris Transport calculation assumes that 3 of the 4 bioshield doors are modified to minimize the potential of debris blockage. The final design of the modification to the doors is not complete.

2. Has the licensee updated its licensing basis to reflect the corrective actions taken in response to GL 2004-02?

   The inspectors verified that changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), that were identified in the licensee's GL 2004-02 response were reviewed and documented in accordance with 10 CFR 50.59 and PSEG had obtained NRC approval prior to implementing those changes that require such approval as stated in 10CFR 50.59. Finally, the inspectors verified that PSEG intends to update the Salem Unit 2 licensing bases to reflect the final modification and associated procedure changes taken in response to GL 2004-02.

The TI will remain open to allow for the review of portions of the GL response that have not been completed. Specifically, PSEG had not completed their downstream effects analysis or chemical precipitant analysis. The results of these analyses have the potential to impact the final size of the strainer, licensing basis and programmatic procedures. Therefore, the inspection will be considered incomplete until the results are reviewed. PSEG plans to evaluate the strainer for adequacy once the test results that quantify the head loss are known. The NRC has set a December 2007 deadline for the completion of these evaluations. PSEG requested and received approval to defer replacement of CalSil insulation on the steam generators and the pressurizer surge line until the next refueling outage, when the steam generators are going to be replaced.

c. <u>Findings</u>

No findings of significance were identified.

.2 Temporary Instruction 2515/169: Mitigating Systems Performance Index Verification

a. Inspection Scope (1 sample)

Temporary Instruction (TI) 2515/169 was issued on July 25, 2006, to verify that licensees have correctly implemented the Mitigating Systems Performance Index (MSPI) guidance for reporting unavailability and unreliability of the monitored safety systems. The MSPI replaces the four Safety System Unavailability (SSU) indicators currently in use in the Reactor Oversight Process. The MSPI monitors the unavailability and unreliability of the same four safety systems that comprise the SSU; however it also monitors the cooling water support systems for those four safety systems. Specifically for Salem, the systems monitored by MSPI are: Emergency Alternating Current (EAC), high pressure injection (HPI), heat removal by auxiliary feedwater (AFW), residual heat removal (RHR), and cooling water support systems represented by service water (SW), and component cooling water (CCW).

The inspectors reviewed the Salem MSPI Bases Document, Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, station procedures, Maintenance Rule data, notifications, control room operating logs, Technical Specification log entries, system engineering notebooks, and compilation of MSPI data to complete the following inspection requirements:

- On a sampling basis, the inspector will review the licensee's list of surveillance activities which, when performed, do not render the train unavailable due to the short duration of the activity (less than 15 minutes).

- On a sampling basis, the inspector will review the licensee's list of surveillance activities which, when performed, do not render the train unavailable due to the credit for operator recovery activities as defined by Nuclear Energy Institute (NEI) 99-02 (Regulatory Assessment Performance Indicator Guideline), Revision 4, page F-6.

- For each MSPI system, using the general concepts discussed in Section 1.2.2 of Appendix F of NEI 99-02, Revision 4, the inspector will independently determine the baseline planned unavailability hours and confirm that these hours were correctly translated into the basis document.

- On a sampling basis for each MSPI system, using operating logs, corrective maintenance records, and condition reports, the inspector will confirm that the actual planned and unplanned unavailability data is accurate.

- On a sampling basis for each MSPI system, based on a review of related maintenance and test history, the inspector will confirm the accuracy of the failure data (demand failures, run/load failures, and failures to meet the risk-significant mission time, as applicable) for the identified monitored components.

The TI requires the documentation of the following questions and their answers:

1. For the sample selected, did the licensee accurately document the baseline planned unavailability hours for the MSPI systems?

2. For the sample selected, did the licensee accurately document the actual unavailability hours for the MSPI systems?

3. For the sample selected, did the licensee accurately document the actual unreliability information for each MSPI monitored component?

4. Did the inspector identify significant errors in the reported data, which resulted in a change to the indicated index color? Describe the actual condition and corrective actions taken by the licensee, including the date when the revised PI information was submitted to the NRC.

5. Did the inspector identify significant discrepancies in the basis document which resulted in (1) a change to the system boundary; (2) an addition of a monitored component; or (3) a change in the reported index color? Describe the actual condition and corrective actions taken by the licensee, including, the date of when the bases document was revised.

b. Findings and Observations

No findings of significance were identified.

In response to the five questions listed in the Inspection Scope, the inspectors determined that:

1. PSEG accurately documented the baseline planned unavailability hours for the MSPI systems.

2. PSEG accurately documented the actual unavailability hours for the MSPI systems.

3. PSEG accurately documented the actual unreliability information for each MSPI monitored component.

4. The inspectors did not identify significant errors in the reported data that resulted in a change to the indicated index color.

5. The inspectors did not identify significant discrepancies in the basis document which resulted in (1) a change to the system boundary; (2) an addition of a monitored component; or (3) a change in the reported index color.

.3 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of the Salem Generating Station conducted in August 2006. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC assessment of PSEG's performance and to verify if any significant safety issues were identified that required further NRC review.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On January 5, 2007, the resident inspectors presented the inspection results to Messrs. W. Levis, T. Joyce and C. Fricker. None of the information reviewed by the inspectors was considered proprietary.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by PSEG and are violations of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCV(s).

- Salem Generating Station Technical Specification 3.7.12 requires that the combination of initial enrichment, burn-up, and Integral Fuel Burnable Absorber (IFBA) of each fuel assembly stored in Region 1 or Region 2, shall be within the acceptable limits described in associated surveillance requirements. Surveillance requirement 4.7.12.2 requires administrative verification that the fuel assemblies meet storage constraints described by enrichment and irradiation (burn-up) prior to storing fuel assemblies in Region 2. Contrary to this requirement, PSEG stored one fuel assembly in Region 2 of the Unit 1 SFP and twelve fuel assemblies in Region 2 of the Unit 2 SFP that did not meet the required limits and constraints for several years. This was identified in PSEG's corrective action program as notifications 20300285 and 20306052. The finding was of very low safety significance because SFP reactivity was acceptably low while this condition existed.

- 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, PSEG did not prescribe procedures to maintain ventilation damper actuator air pressure within design limits. This was identified in PSEG's corrective action program as notification

20298713. The finding was of very low safety significance because there was no actual loss of safety function and the initiating event likelihood of a steam line break in the 13 AFW pump enclosure is very small.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

T. Joyce, Salem Vice President
C. Fricker, Salem Plant Manager
S. Robitski, Salem Engineering Director
J. Stone, Salem Maintenance Director
G. Sosson, Salem System Engineering Manager
R. Gary, Salem Technical Superintendent - Radiation Protection
S. Mannon, Salem Regulatory Assurance Manager
M. Gwirtz, Assistant Operations Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened/Closed

05000272&311/2006005-01	NCV	Inadequate Procedure Implementation for Scaffold Construction (Section 1R04)
05000311/2006005-02	NCV	Failure to Implement Effective Corrective Actions for Reactor Coolant System Tubing Leak (Section 1R08)
05000311/2006005-03	NCV	Incorrectly Positioned Fuel Assembly (Section 1R20)

Opened/Closed

05000311/2006003-00	LER	Manual Reactor Trip due to Elevated Reactor Coolant Pump Seal Leak-off (Section 4OA3.1)
05000272/2006002-00	LER	Inoperability of the Auxiliary Feedwater System Due to a Partially Opened Damper (Section 4OA3.2)
05000311/2006004-00	LER	Fuel Assemblies Misloaded in the Spent Fuel Pool Racks (Section 4OA3.3)
05000311/2006005-00	LER	Automatic Start of Auxiliary Feedwater Pumps in Mode 4 (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

In addition to the documents identified in the body of this report, the inspectors reviewed the following documents and records:

Section 1R01: Adverse Weather Protection

Procedures
SC.OP-SO.SA-0002, Temporary Station Air compressor Operation, Rev. 16
SC.MD-GP.ZZ-0001, Station Preparations for Winter - Mechanical, Rev. 6
SC.MD-GP.ZZ-0178, Station Preparations for Winter - Electrical, Rev. 13
SH.OP-DG.ZZ-0011, Station Seasonal Readiness Guide, Rev. 5
SC.OP-PT.ZZ-0002, Station Preparations for Seasonal Conditions, Rev. 10
WC-AA-107, Seasonal Readiness, Rev. 2
OP-AA-108-111-1001, Severe Weather and Natural Disaster Guidelines, Rev. 2
NC.OP-DG.ZZ-0002, Severe Weather Guide, Rev. 6
SC.OP-AB.ZZ-0001, Adverse Environmental Conditions, Rev. 9

Notifications

20306611	20306200	20306245	20304086
20306612	20306397	20304711	20306397
20308212	20264280	20264311	

Orders

30132532	30132299	CROD 05-022

Section 1R04: Equipment Alignment

Procedures
S2.OP-SO.SW-0003, 22 Nuclear Service Water Header Outage
SH.MD-DG.ZZ-0023, Scaffold Erection, Modification and Dismantling Desk Top Guide, Rev. 6
SH.MD-AP.ZZ-0023, Scaffold Program, Rev. 8
S2.OP-ST.SW-0013, Service Water Valve Verification Modes 1-4, Rev. 1
S2.OP-AB.SW-0001, Loss of Service Water Header Pressure, Rev. 14
S2.OP-AB.SW-0002, Loss of Service Water - Turbine Header, Rev. 11
S2.OP-AB.SW-0003, Service Water Bay Leak, Rev. 7
S2.OP-SO.SW-0001, Service Water Pump Operation, Rev. 21
S2.OP-SO.SW-0005, Service Water System Operation, Rev. 36
S2.OP-IO.ZZ-0010, Spent Fuel Pool Manipulations, Rev. 23
S2.OP-SO.SF-0002, Spent Fuel Pool Cooling System Operation, Rev. 17
S2.OP-SO.SF-0006, Spent Fuel Pool Emergency Fill, Rev. 7

Drawings
205342, Rev. 71, No. 2 Unit Service Water Nuclear Area
205342 SH 1-7, Revs. 73-72-71-57-63-63-06 (respectively), No. 2 Unit Service Water Nuclear Area

205342, 205322, 205333, 205337, 205342

Completed Surveillances
S2.OP-ST.SW-0013, Service Water Valve Verification Modes 1-4, dated 9/9/06 & 10/5/06

Evaluations
70062702

Notifications

20124408	20268335	20300122	20302846
20188602	20270769	20300135	20302903
20202561	20272842	20300171	20302985
20227177	20274738	20300380	20303234
20227287	20283168	20301414	20303540
20249645	20283841	20302520	20303542
20263085	20285934	20302539	20303575
20263086	20299815	20302642	20300045
20264006			

Work Orders

60020422	60054333	60059597	60060640
60048252	60057743	60060302	60062921
60054330	60057916	60060322	60063285

Other Documents
WCDs 4175660, 4175662, 4175665, 4175666
SGS Unit 2 PRA Risk Evaluation Form for Work Week No. 1448 (10/29/06 - 11/04/06)
SGS DCP Status Report, dated 10/31/06
Salem Operations Concern List, dated 11/03/06
Salem Operations Work Around List, dated 11/03/06
Salem Unit 2 Outage List, dated 10/31/06
Salem Control Room Distractions List, dated 10/30/06
Salem Non-Outage Elective Maintenance Backlog, dated 10/30/06
Salem Unit 2 SW Mechanical TRIS Lineup, dated 10/31/06
Risk-Informed Inspection Notebook for Salem Generating Station, Revision 2
S-C-SF-MEE-1103, Spent Fuel Pool Cooling System: Seismic Upgrade to Preclude Pool Boiling, Rev. 2
Tagging Work List 4177220, 2B Diesel Generator

Section 1R05: Fire Protection

Procedures
FRS-II-711, Pre-Fire Plan U1 & U2 Fuel Handling Building, Rev 2
FRS-II-421, Pre-Fire Plan U1 & U2 Fuel Handling Building, Rev 2
FRS-II-521, Pre-Fire Plan U1 & U2 Inner Piping Penetration Area & Chiller Rooms
FRS-II-441, Pre-Fire Plan U1 & U2 Relay and Battery Rooms, and Corridor
FRS-II-411, Pre-Fire Plan U1 & U2 Reactor Plant Auxiliary Equipment Area and
SC.FP-AP.ZZ-0003, Actions for Inoperable Fire Protection - Salem Station, Rev. 11

FRS-II-611, Pre-Fire Plan U1 & U2 Reactor Containment, Rev. 5
NC.FP-AP.ZZ.0020, Compensatory Measure Firewatch Program, Rev. 1
SC.FP-SV.FBR-0026(Q), Flood and Fire Barrier Penetration Seal Inspection, Rev 3

Notifications

20301354	20298296	20297881	20295591

Section 1R06: Flood Protection Measures

Procedures
SC.FP-SV.FBR-0026(Q), Flood and Fire Barrier Penetration Seal Inspection, Rev 3
ND.DE-PS.ZZ-0010(Q)-A5, Internal Hazards Program Flooding Analysis Methodology, dated 2/8/95

Drawings

205226	205326

Notifications

20304038	20304039	20304040

Other Documents
Updated Final Safety Analysis Report section 3.6.5
Salem Generating Station Individual Plant Examination, Section 3.3.8 Internal Flooding, dated July 1993
Salem Generating Station Probalistic Risk Assessment, Section 3.10.2 Internal Flooding Analysis, dated August 1998
NRC Information Notice 2005-11, Internal Flooding/Spray-Down of Safety Related Equipment Due to Unsealed Equipment Hatch Floor Plugs and/or Blocked Floor Drains

Section 1R07: Heat Sink Performance

Procedures
S2.OP-PT.SW-0027, 22 Component Cooling Heat Exchanger Heat Transfer Performance Data Collection, Rev. 12

Orders

30065642	30124460

Section 1R08: Inservice Inspection Activities

Procedures
SC.OP-AP.ZZ-0108(Q) - Revision 11, 5/1/06; Operability Assessment And Equipment Control Program
ER-AA-335-002, Revision 3, Liquid Penetrant Examination
SH.RA-IS.ZZ-0005(Q), Revision 6, 7/18/05; VT-2 Visual Examination Of Nuclear Class 1, 2, and 3 Systems
Framatome ANP, Inc. Nondestructive Examination Procedure, 54-ISI-187-12, Revision 12, 7/22/04; Ultrasonic Examination Of Reactor Vessel Flange To Shell Welds From Flange Top Surface
Framatome ANP, Inc. Nondestructive Examination Procedure, 54-ISI-132-08, Revision 8, 8/15/05; Manual Ultrasonic Examination of Vessel Nozzle Inner Radius Regions
Areva, NP Inc. Nondestructive Examination Procedure, 54-ISI-836-10, Revision 10, 7/12/06; Ultrasonic Examination of Austenitic Piping Welds
Areva, NP Inc. Nondestructive Examination Procedure, 54-ISI-135-07, Revision date, 1/24/06; Linearity and Beam Spread Measurements
SH.RA - AP.ZZ - 8805(Q) - Revision 4, 8/31/06; Boric Acid Corrosion Management Program
ER - AP - 331, Revision 3, Boric Acid Corrosion Control (BACC) Program
ER - AP - 331 - 1001, Revision 2, Boric Acid Corrosion Control (BACC) Inspection Locations, Implementation And inspection Guidelines
ER - AP - 331 - 1002, Revision 3, Boric Acid Corrosion Control (BACC) Program Identification, Screening, and Evaluation
ER - AP - 331 - 1003, Revision 1, RCS Leakage Monitoring And Action Plan
ER - AP - 331 - 1004, Revision 2, Boric Acid Corrosion Control (BACC) Program Training And Qualification
LS - AA - 125, Revision 10; Corrective Action Program (CAP) Procedure
LS - AA - 120, Revision 5; Issue Identification And Screening Process

Drawings/Isometrics
PSE&G Nuclear Dwg. 205301 A 8762-31, Sh. 2, 5/10/05
PSE&G Nuclear Dwg. 239093 B 9639-16, 2, 5/10/05
Westinghouse Dwg. E 234-456-8, Revision 8, 5/21/71; Instrumentation Penetration Assy. Abd Details - Bottom Head
PSE&G Nuclear Dwg. 207072 A 8797-14; Salem Nuclear Generating Station U2 - Reactor Containment Floor Plan El 130' - 0"
PSE&G Nuclear Dwg. 207071 A 8797-5; Salem Nuclear Generating Station U2 - Reactor Containment Floor Plan El 100' - 0"
PSE&G Nuclear Dwg. 207070 A 8797-8; Salem Nuclear Generating Station U2 - Reactor Containment Floor Plan El 78' - 0" & 81' - 0"

Notifications/Condition Report:

20301356	20301788	20260483	20274056
20301357	20301668	20256369	20274069
20301358	20301212	20256578	20260485
20301359	20274037	20254499	20253340
20301355	20274065	20272954	20256101
20301922	20258243	20274063	20257370

20260486
20256315
20256316
20257196
20273978
20282812
20262983
20265089
20269832
20269833
20269834
20273648
20302225
20299931
20299933
20299912
20299901
20299840
20299890
20300060
20300105
20300125
20300126
20300130
20300220
20300221
20300222
20300230
20301409
20261137
20263672
20270168
20300263
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20300313
20300320
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20300329
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20300525
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20300525
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20302452
20301734
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20301117
20266701
20285799
20286336
20260484
20300614
20300567
20300568
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20300634
20300651
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20300657
20300768
20299708
20300081
20300083
20300927
20301408
20301118
20301119
20301120
20301121
20301122
20308078
20300977
20263664
20260628
20256263
20257880

<u>NDE Examination Reports & Data Sheets</u>
20300090, PT
20298612, PT
20299618, PT
002701, UT-05-146
002701, UT-05-147
002701, UT-05-148
BS-05-001
BS-05-002
BS-05-003
011200, UT-05-161
011200, UT-05-159
011200, UT-05-160
011200, UT-05-162
011500, UT- 05-158
011500, UT-05-155
011500, UT-05-156
012510, VT2 Pressurizer Heater Penetrations
004602, VT2 29-RC-1230-1-BMV, Nozzle to Safe End

004612, VT2 29-RC-1240-1-BMV, Nozzle to Safe End
020300, UT-05-165
020300, UT-05-164
020300, UT-05-163
050455, PT-05-013
161200, UT-05-128
161200, UT-05-129
165100, UT-05-134
165100, UT-05-135
165100, UT-05-136
183550, PT-05-011
183700, PT-05-012
355500, UT-05-016
355500, UT-05-017
005200, UT-05-090
005200, UT-05-091
005200, UT-05-092
Wesdyne Final Inspection Report: Reactor Coolant Pump Shaft #21 (ID 979-933D303), 10/19/06

Repair-Replacement Work Orders

60065645	30000289	30095917	80086664
60004071	60047571	60059601	60060308
30096046	60060725		

Calculations/Evaluations
S-2-RC-MEE-1983, Revision 0, 12/06; 2R15 Steam Generator Degradation Assessment For Salem Unit 2
70063151
70062527

Miscellaneous Documents
Non-Routine/Non-Scheduled Chemistry Sample Analysis, 10/12/06; Walkway Ceiling Crack Between 23 & 24 RCP (in bioshield)
Non-Routine/Non-Scheduled Chemistry Sample Analysis, 10/12/06; 2CC302 Piping in letdown Heat Exchanger Room

Section 1R11: Licensed Operator Requalification Program

Other Documents
Scenario SG-0681, “Loss of 2A 4KV Bus, AB.RCP, Trip- ½ with Subsequent Steam Break”

Section 1R12: Maintenance Effectiveness

Procedures
ER-AA-310, Implementation of the Maintenance Rule, Rev 5
ER-AA-310-1003, Maintenance Rule - Performance Criteria Selection, Rev 3
ER-AA-310-1004, Maintenance Rule - Performance Monitoring, Rev 5
ER-AA-310-1005, Maintenance Rule - Dispositioning Between (a)(1) and (a)(2), Rev 3
ER-AA-310-1008, Exelon Maintenance Rule Process Map, Rev 0
SC.MD-PM.ZZ-0135, Ventilation Damper Inspection and Guidelines, Rev. 6

Drawings

205216	610662	205337
205242	252506	236114

Notifications

20302395	20265955	20284716	20230869
20251103	20243881	20272719	20290907
20265802	20240185	20223898	20291358
20265957	20285064	20223623	20305609
20290886	20304397	20305194	20299045
20288156	20270113	20266000	20263512
20262604	20304496	20299155	20298927
20307496	20295477	20218748	20298070

Orders

70039685	60057070	60063939	70061862
70044777	70039685	70054731	70062026
		70059410	70062027

Other Documents
Updated Final Safety Analysis Report, Section 9.4, Heating, Ventilation, and Air Conditioning Systems
12 Chiller Unavailability (Cumulative)
12 Chiller Unavailability (Monthly)
SAEP 2002-06, “Salem Expert Panel Meeting Minutes, 6/12/2002"
Salem Unit 1 Narrative Logs
PIRS 970616184
SAEP 2006-05, Salem Maintenance Rule Expert Panel Minutes, meeting date May 31, 2006
MPR Associates, Inc., Past Operability of the Salem Unit 1 Auxiliary Feedwater Pumps, dated November 27, 2006
Salem Generating Station Licensee Event Report 2006-002, Inoperability of the Auxiliary Feedwater System Due to a Partially Opened Damper, dated November 27, 2006
Quarterly SHIP System Report for Unit 1 Auxiliary Building Ventilation
Quarterly SHIP System Report for Unit 2 Auxiliary Building Ventilation
VTD 320766, Fabco Fastening Systems, Rev. 3

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Completed Surveillances
S2.OP-ST.SW-0001, Inservice Testing - 21 Service Water Pump, dated 8/21/06
S2.OP-ST.SW-0002, Inservice Testing - 22 Service Water Pump, dated 9/27/06
S2.OP-ST.SW-0003, Inservice Testing - 23 Service Water Pump, dated 9/11/06
S2.OP-ST.DG-0001, 2A Diesel Generator Surveillance Test, dated 10/01/06
S2.OP-ST.DG-0002, 2B Diesel Generator Surveillance Test, dated 9/22/06
S1.OP-PT.SW-0017, 12 Component Cooling Heat Exchanger Heat Transfer Performance Data Collection, Rev. 14

Procedures
SC.OP-AB.SW-0008, Rev. 3, Service Water Test-Line Cross-Connect Alignment
S2.OP-AB.SW-0004, Rev. 5, Loss of Service Water During Service Water Header Outage
SC.OM-AP.ZZ-0001, Rev. 1, Shutdown Safety Management Program - Salem Annex
OU-AA-103, Shutdown Safety Management Program, Rev. 6
S2.OP-SO.SW-0003, Rev. 21, 22 Nuclear Service Water Header Outage
S1.OP-SO.CC-0001, Component Cooling System Operation, Rev. 16
S1.OP-SO.CC-0002, 11 & 12 Component Cooling Heat Exchanger Operation, Rev. 24
S2.OP-SO.RC-0006, Rev. 28, Draining The Reactor Coolant System <101Foot Elevation With Fuel in The Vessel
S2.OP-SO.RC-0005, Rev. 33, Draining The Reactor Coolant System To ≥ 101 Foot Elevation

Drawings

205231	205242

Notifications

20299728	20300069	20300441	20300097
20299841	20300075	20297844	

Orders
30103201

Other Documents
Risk-Informed Inspection Notebook For Salem Generating Station, Rev. 2
S-2-ZZ-RZZ-0034, Rev. 6, ORAM Safety Functional Assessment Trees (SFATs) for Salem Unit 2
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PSEG S2R15 Refueling Outage Risk Assessment, dated 8/16/06
WCDs 4175660, 4175662, 4175665, 4175666, 4189660
SOD-2006-058, Mid-loop Operations in Support of Vacuum Fill Activity, dtd October 23, 2006
S-1-CC-MDC-1817, Component Cooling System Thermal-Hydraulic Analysis - Unit 1, Rev. 4
SGS Unit 1 PRA Risk Evaluation Form for the week of November 26, 2006
Contingency Plan for Inventory Control and Shutdown Cooling

Section 1R15: Operability Evaluations

Procedures
SH.OP-AP.ZZ-0108, Operability Assessment and Equipment Control Program, Rev. 24
S2.OP-ST.CVC-0004, Inservice Testing 22 Charging Pump, Rev 20
S2.OP-ST.NIS-0002, Power Distribution - Quadrant Power Tilt Ration, Rev. 11
SC.RE-ST.ZZ-0004, Quadrant Power Tilt Ratio, Rev. 1
SC.RE-ST.ZZ-0015, Quadrant Power Tilt Ratio Measurement, Rev. 0
SC.IC-FT.NIS-0011, N31 Source Range (In Shutdown), Rev. 1
SC.IC-FT.NIS-0012, N32 Source Range (In Shutdown), Rev. 2
S2.IC-GP.NIS-0001, Nuclear Instrumentation System Data Procedure, Working Copy

Notifications

20281911	20268567	20303419	20300677
20300676	20301991	20303770	

Orders

70053276	30122727	80046667	80017036
30122829	50086097	80087996	
50086087	70062710		

Other Documents
VTD 301137 - Instructions for Installing, Operating and Maintaining Manual 2286 Pacific Pump
VTD 326397 - CVC Charging Pump Lube Oil Piping

Section 1R17: Permanent Plant Modifications

Drawings

606076	80057520
	SUP03R0

Notifications

20305347	20302683	20307951	20307891
20303591	20308058	20307962	

Orders
80057587

Other Documents
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SC-RM010-01, Salem Unit 2 Main Steam Line Radiation Monitor 2R46A/B/C/D/E, Rev. 0
NUREG-0737, Clarification of TMI Action Plan Requirements, dated November 1980
Salem Generating Station Special Report 311/06-002, Main Steam Line Discharge Monitors (R46s) Inoperable Greater than Seven (7) Days, dated October 2, 2006

Salem Generating Station Response to RAIs on LCR S06-03, Request for Change to Technical Specifications Accident Monitoring Instrumentation and Source Check Definition, dated October 9, 2006

Section 1R19: Post-Maintenance Testing

Procedures

S2.OP-ST.CBV-0003, Containment Systems - Cooling Systems, Rev. 14
S2.OP-ST.SW-0010, Inservice Testing Containment Fan Cooler Unit (CFCU) Service Water Valves, Rev. 19
S2.RA-ST.SW-0010, Inservice Testing Containment Fan Cooler Unit (CFCU) Service Water Valves Acceptance Criteria, Rev. 38
SH.MD-GP.ZZ-0240, System Pressure Test at Normal Operating Pressure and Temperature, Rev. 7
SC.MD-CM.RC-0001, Reactor Coolant Pump Seal Disassembly, Inspection, Repair and Assembly, Rev. 25
S2.OP-PT.CAN-0001, Containment Walkdown, Rev. 17
S2.OP-ST.RC-0007, Seal Injection Flow, Rev. 6
S2.OP-ST.CVC-0004, Inservice Testing - 22 Charging Pump, Rev. 20
SC.MD-PM.ZZ-0052, Disassembly, Inspection and Reassembly of Velan Swing Check Valves Mark #s A-160, A-224, AA-64, AA-121, AA-122, AA-153, E-6, EA-8, FA-33 and FA-34, Rev. 5;
S2.OP-ST.CVC-0003, Inservice Testing - 21 Charging Pump, Rev. 19
S2.OP-ST.CS-0001, Inservice Testing - 21 Containment Spray Pump, Rev. 18
S2.OP-ST.DG-0008, 2C Diesel Generator Auxiliaries - Air Start Valve Test, Rev. 10
S2.OP-PT.DG-0018, 2C Diesel Generator Engine Lube Oil Header Low Pressure Trip and Overspeed Trip Functional Test, Rev. 4
S2.RA-ST.CS-0001, Inservice Testing 21 Containment Spray Pump Acceptance Criteria, Rev. 6
S2.RA-IS.ZZ-0013, Reactor Containment Building Integrated Leak Rate Test, Rev. 2
SH.MD-GP.ZZ-0240, System Pressure Test at Normal Operating Pressure and Temperature, Rev. 7
Test Plan # 80083925, #3 Station Air Compressor Site Acceptance Test, Rev. 3

Drawings

205335	233430	218215

Notifications

20302853	20303064	20302967	20304121
20302842	20306803	20300863	20305225
20305249	20305253		

Orders

30099095	70061392	50097908	60056764
30033947		60065649	
50099152		60056577	
80075675		80083925	
70059937		70060167	

60066003	70063242
30102532	60050018
60066694	30130351
70059949	70059935
70059905	70061228
80090205	

Other Documents
S-C-CBV-MDC-1637, Containment Fan Cooler Unit Design Basis Capacity, Rev. 2
ACM 06-034, 21 RCP Elevated Vibrations Adverse Condition Monitoring
SER OTDM 06-029, 21 Reactor Coolant Pump (S2RC-2RCE3)
Technical Evaluation 70063242-0010, 21 RCP Elevated Vibration
Vendor Technical Document (VTD) 106250, Velan Swing Check Valve
DE-CB.CA-0014, Configuration Baseline Documentation for Control Air and Station Air Systems, Rev. 5

Section 1R20: Refueling and Outage Activities

Procedures
S2.OP-SO.RC-0005, Draining the Reactor Coolant System to ≥101 foot Elevation, Rev. 33
S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load, Rev. 26
S2.OP-SO.MS-0002, Steam Dump System Operation, Rev. 12

Notifications

20301199	20302814	20302842	20304938
20302815	20302353	20302748	20304948
20302822	20302163	20300275	20304970
20300441	20300097	20300069	20303850
			20303443

Other Documents
Contingency Plan for Inventory Control and Shutdown Cooling

Section 1R22: Surveillance Testing

Procedures
S2.OP-ST.SSP-0002, SEC Mode Ops Testing 2A Vital Bus, Rev. 28
S2.OP-ST.RC-0008, Reactor Coolant System Water Inventory Balance, Rev. 27
S1.OP-ST.SJ-0001, Inservice Testing - 11 Safety Injection Pump, Rev. 15
S1.OP-RA.SJ-0001, Inservice Testing -11 Safety Injection Pump Acceptance Criteria, Rev. 6
S1.OP-ST.SW-0005, Inservice Testing - 15 Service Water Pump, Rev. 23
S1.RA-ST.SW-0005, Inservice Testing - 15 Service Water Pump Acceptance Criteria, Rev. 8
S1.OP-ST.RC-0008, Reactor Coolant System Water Inventory Balance, Rev. 19

Orders

50086489	50097618

Section 1R23: Temporary Plant Modifications

Procedures
NC.DE-AP.ZZ-0030, Control of Temporary Modifications

Orders
70062259

Section 2OS1: Access Control to Radiologically Significant Areas

Notifications

20282851	20283233	20283707	20283719
20284066	20284136	20284240	20285539
20286642	20286824	20287142	20288549
20288560	20288606	20288624	20289159
20289436	20291477	20292367	20293786
20298001	20298306	20298559	20298591
			20300088

Section 2OS2: ALARA Planning and Controls

Other Documents
ALARA Plan: No. 63, Containment Sump DCP
ALARA Plan: No. 54, Nozzle Dam/Bowl Work Chanel Head Entries
ALARA Plan: No. 50, 2R15 Scaffolding Activities U/2 Containment
ALARA Plan: No. 58, 21-24 SG SSI/FOSAR
ALARA Plan: No. 55, SG Primary Eddy Current
ALARA Plan: No. 104, Set and Demob SG's

Section 4OA2: Identification and Resolution of Problems

Procedures
1-EOP-LOPA-1, Loss of All AC Power, Rev. 23
1-EOP-APPX-6, RCP Seal Cooling Restoration, Rev. 22
2-EOP-APPX-6, RCP Seal Cooling Restoration, Rev. 22
2-EOP-LOPA-1, Loss of All AC Power, Rev. 24
2-EOP-LOPA-2, Loss of All AC Power Recovery / SI Not Required, Rev. 22
2-EOP-LOPA-3, Loss of All AC Power Recovery / SI Required, Rev. 23
S2.OP-AB.CR-002(Q), Control room Evacuation Due to Fire in the Control Room, Relay room, 460/230V Switchgear room or 4KV Switchgear room, Rev 22
NC.IS-TM.ZZ-0001(Z), Nuclear Department Safety Manual, Rev 8

Calculations/Analyses
314204(1), Station Blackout Analyses, Dated 7/29/91
S-C-ABV-MDC-1881, Salem Units 1 and 2 ABV Gothic Appendix R Scenarios", Rev 1
S-C-AF-MDC-0727, Condensate Inventory for Decay Heat Removal During Station Blackout, Revision 1

SC-AF002-02, Salem Unit 1&2 AFW Pumps Suction Pressure Low Setpoint Tornado Armed Trip, Rev 0

Notifications

20277247	20271973	20272356	20277379
20286593	20293561	20284189	20299672
20288337	20295458	20284602	20300285
20289619	20303539	20285772	20305884
20290182	20283569	20285771	20306097
20292024	20283840	20287257	
20283861	20283838	20287125	

Orders

70055676	70054164	70055425	70054295

Other Documents
Purchase Order 4500358141
Bases Document 2-EOP-LOPA-1, Rev 25

Section 4OA3: Event Followup

Procedures
S2.OP-ST.TRB-0002, Turbine Protection System Full Functional Test, Rev. 20
S2.OP-ST.MS-0003, Steam Line Isolation and Response Time Testing, Rev. 16
S2.OP-AB.RCP-0001, Reactor Coolant Pump Abnormality, Rev. 19
SC.OP-DG.ZZ-0101, Salem Post-Trip Data Collection Guidelines, Rev. 8
S2.OP-SO.CVC-0001, Charging, Letdown, and Seal Injection,Rev. 27
S2.OP-SO.CVC-0012, CVCS Demineralizer - Normal Operation,Rev. 26
S2.OP-IO.ZZ-0003, Hot Standby to Minmimum Load, Rev. 26

Notifications

20302979	20298382	20298412	20306052
20300402	20298384	20298441	20300285
20298370	20298395	20306261	20306023

Orders

70062027	70063094	50086422	70061462
70060153	30096647		

Other Documents
Adverse Condition Monitoring Plan, Monitoring RCP seal leak off flows during low RCS boron concentration conditions
Salem Lesson Plan NOS05PZRPRT-02 Pressurizer and Pressurizer Relief Tank
Millstone Condition ReportM3-01-0088, Reactor Coolant Pumps B C D Seal Leakoff
Exelon Fuels Department Analysis of Unit 2 Spent Fuel Pool Reactivity Conditions

Section 4OA5: Other Activities

Procedures
2-EOP-LOCA-3, Transfer to Cold Leg Recirculation (Mark-up), Rev. 26
2-EOP-LOCA-5, Loss of Emergency Recirculation (Mark-up), Rev. 24

Drawings
900223, ECCS Emergency Sump Strainer Construction Drawings, Rev. 6
103.132.220.500 Sheet 13, Module 10 Pockets
103.132.270.500 Sheet 2, Containment Layout Salem Unit 2
NKP-SAL2-RC-BK-1, Hot Leg #21 Nukon Blankets Layout, Rev. 0
NKP-SAL2-RC-BK-2, Hot Leg #22 Nukon Blankets Layout, Rev. 0
NKP-SAL2-RC-BK-3, Hot Leg #23 Nukon Blankets Layout, Rev. 0
NKP-SAL2-RC-BK-4, Hot Leg #24 Nukon Blankets Layout, Rev. 0
RM-49528-BD-01, Steam Generator 21 Blowdown, Rev. 2
RM-49528-BD-02, Steam Generator 22 Blowdown, Rev. 2
RM-49528-BD-03, Steam Generator 23 Blowdown, Rev. 2
RM-49528-BD-04, Steam Generator 24 Blowdown, Rev. 2

Notifications (* indicates that this CR generated as a result of this inspection)

20301719	20294989	20251471	20307508
20157540	20297831	20297378	20250868
20161112	20299913	20183687	20307497
20143502	20267520	20140012	20306684
20185873	20253345	20306731	20305450
20253345	20297076	20260710	20302353*
20265377	20175195	20285874	

Orders

60036388	70042152	70055356	70042942
70042942	70041672	70050941	70036348
70036522	70046879	70051642	70063865
70031717	70043865	70050181	70036881
70037213	70050582		80090917
70041840			

Calculations
3 SA-096.020, Structural Analysis of Strainer and Support Structure, Rev. 4
3 SA-096.038, Head Loss Calculation for Present Design Basis, Rev. 2
S-C-RHR-MDC-1711, Available NPSH at RHR Pump in Recirc. Mode, Rev. 3IR1
S-C-RHR-MDC-2039, Debris Generation due to LOCA within Containment for Resolution of GSI-191, Rev. 12
S-C-RHR-MDC-2056, Post-LOCA Debris Transport to Containment Sump for Resolution of GSI-191, Rev. 0IR1
S-C-RHR-MDC-2089, Long Term Wear Effects Evaluation in Support of Resolution of GSI-191, Rev. 0IR0

Miscellaneous

80080788, Salem 2 Containment Sump Upgrades, Rev. 2, dated October 10, 2006
80089513, GSI-191 Insulation Debris Reduction, Rev. 0
80067521 Op 1580, Upgrade Lesson Plan
80080788 Op 0990, Update Design Change Process, Rev. 2, dated October 10, 2006
80089323 OP 0418, Analyze DCP for changes to lesson plans
CC-AA-102, Design Input and Configuration Change Impact Screening, Rev. 13
LR-N05-0103, PSEG Letter: 90-Day Response to Generic Letter 2004-02 “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors”, dated March 4, 2005
LR-N05-0401, PSEG Letter: Response to Generic Letter 2004-02 “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors”, dated September 1, 2005
LR-N06-0253, PSEG Letter: Updated Response to Generic Letter 2004-02 and Request for Extension for Insulation Replacement, dated June 7, 2006
SCN 06-009, UFSAR Change Notice
VS-2006-013, Certificate of Surveillance, dated October 6, 2006
50.59 Screen: GSI-191 Insulation Debris Reduction, Rev. 0
50.59 Screen: Salem 2 Containment Sump Upgrades, Rev. 2
USNRC Letter: Salem Nuclear Generating Station, Unit No. 2 - Approval of Generic Letter 2004-02 Extension Request, dated August 11, 2006
USNRC Letter: Generic Letter 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors", Extension Request Approval for Salem, Unit 2, dated July 7, 2006

LIST OF ACRONYMS

ABV	Auxiliary Building Ventilation
AFW	Auxiliary Feedwater
ALARA	As Low As is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CC	Component Cooling
CCW	Component Cooling Water
CEDE	Committed Effective Dose Equivalent
CFCU	Containment Fan Coil Unit
CFR	Code of Federal Regulations
CR	Condition Report/Notification
DCP	Design Change Package
EAC	Emergency Alternating Current
EDG	Emergency Diesel Generator
ESF	Engineered Safeguards Feature
GL	Generic Letter
GPM	Gallons Per Minute
GSI	Generic Safety Issue
HPI	High Pressure Injection
ICA	Item Control Area

IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
ISI	Inservice Inspection
LDE	Lens Dose Equivalent
LER	Licensee Event Report
LHP	Lower Head Penetration
MDAFW	Motor Driven Auxiliary Feedwater
MSIV	Main Steam Isolation Valve
MSPI	Mitigating Systems Performance Index
MT	Magnetic Particle Testing
NCV	Non-cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NOTF	Notification
NRC	Nuclear Regulatory Commission
ORAM	Outage Risk Assessment and Management
PI	Performance Indicator
PRA	Probabilistic Risk Assessment
PSEG	Public Service Enterprise Group Nuclear LLC
PT	Penetrant Testing
PWR	Pressurized Water Reactor
QHPI	Quick Human Performance Investigation
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RE	Reactor Engineer
RFO	Refueling Outage
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
S2R15	Fifteenth Refueling Outage of Salem Unit 2
SBO	Station Blackout
SCWE	Safety Conscious Work Environment
SDE	Skin Dose Equivalent
SDP	Significance Determination Process
SFAT	Safety Functional Assessment Tree
SFP	Spent Fuel Pool
SGS	Salem Generating Station
SPT	Station Power Transformer
SRO	Senior Reactor Operator
SSU	Safety System Unavailability
SW	Service Water
TEDE	Total Effective Dose Equivalent
TI	Temporary Instruction
TRIS	Tagging Request Information System
TS	Technical Specification
TSO	Transmission System Operator
TWS	Traveling Water Screen

UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VT	Visual Examination
WCD	Work Clearance Document
WO	Work Order