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

August 5, 2008

Mr. William Levis
President and Chief Nuclear Officer
PSEG Nuclear LLC
80 Park Plaza, T4B
Newark, NJ 07102

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - NRC INTEGRATED INSPECTION REPORT 05000272/2008003 and 05000311/2008003

Dear Mr. Levis:

On June 30, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Salem Nuclear Generating Station, Unit Nos. 1 and 2. The enclosed integrated inspection report documents the inspection results discussed on July 9, 2008, with Mr. Braun 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 three self-revealing findings of very low safety significance (Green). These findings were all determined to involve violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) 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, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Salem Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be 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/

Arthur L. Burritt, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000272/2008003 and 05000311/2008003
w/Attachment: Supplemental Information

cc w/encl:
T. Joyce, Senior Vice President, Operations
R. Braun, Site Vice President
K. Chambliss, Director, Nuclear Oversight
J. Spears, Acting Director of Finance
G. Gellrich, Salem Plant Manager
J. Keenan, General Solicitor, PSEG
M. Wetterhahn, Esquire, Winston and Strawn, LLP
L. Peterson, Chief of Police and Emergency Management Coordinator
P. Baldauf, Assistant Director, NJ Radiation Protection Programs
P. Mulligan, Chief, NJ Bureau of Nuclear Engineering, DEP
H. Otto, Ph.D., Administrator, DE Interagency Programs, DNREC Div of Water Resources
Consumer Advocate, Office of Consumer Advocate, Commonwealth of Pennsylvania
N. Cohen, Coordinator - Unplug Salem Campaign
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

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/
Arthur L. Burritt, Chief
Projects Branch 3
Division of Reactor Projects

Distribution
S. Collins, RA
M. Dapas, DRA
D. Lew, DRP
J. Clifford, DRP
A. Burritt, DRP
L. Cline, DRP
J. Bream, DRP
D. Schroeder, DRP, Sr Resident Inspector
H. Balian, DRP, Resident Inspector
K. Venuto, DRP, Resident OA
S. Williams, RI OEDO
R. Nelson, NRR
H. Chernoff, NRR
R. Ennis, NRR, PM
J. Shea, NRR, Backup
D. Collins, NRR
S. Bailey, NRR
T. Valentine, NRR
ROPreports Resource
Region I Docket Room (with concurrences)

DOCUMENT NAME: G:\DRP\BRANCH3\INSPECTION\REPORTS**ISSUED**\SAL0803.DOC

ML082200302

SUNSI Review Complete: ALB (Reviewer's Initials)

After declaring this document "An Official Agency Record" it **will** be released to the Public.

To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP		RI/DRP			
NAME	DSchroeder/ ALB for		LCline/ JRB for		ABurritt/ ALB			
DATE	08/ 06 /08		08 / 07 /08		08/ 06 /08			

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	50-272, 50-311
License Nos:	DPR-70, DPR-75
Report No:	05000272/2008003 and 05000311/2008003
Licensee:	PSEG Nuclear LLC (PSEG)
Facility:	Salem Nuclear Generating Station, Unit Nos. 1 and 2
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	April 1, 2008 through June 30, 2008
Inspectors:	D. Schroeder, Senior Resident Inspector H. Balian, Resident Inspector J. Orr, Senior Project Engineer D. Everhart, Reactor Inspector J. Furia, Senior Health Physicist P. Kaufman, Senior Reactor Inspector T. O'Hara, Reactor Inspector A. Ziedonis, Reactor Inspector
Approved By:	Arthur L. Burritt, Chief Projects Branch 3 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000272/2008003, 05000311/2008003; 04/01/2008 – 06/30/2008; Salem Nuclear Generating Station Unit Nos. 1 and 2; Maintenance Risk Assessment and Emergent Work Control, Operability Evaluations, Surveillance Testing.

The report covered a three-month period of inspection by resident inspectors, and an announced inspection by a regional radiation specialist, a regional reactor safety inspector, a regional security inspector, and a regional projects inspector. Three Green non-cited violations (NCVs), and one unresolved item 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing non-cited violation of Technical Specification (TS) 6.8.1.a, "Procedures and Programs," was identified because PSEG did not maintain adequate control of the system configuration for the Unit 2 chill water system during maintenance on the 21 chiller. Specifically, on May 27, 2008, all three Unit 2 chill water system chillers tripped due to an error in the safety tagging sequence specified by the work control documents for maintenance on the 21 chiller.

  This finding is more than minor because it is associated with the configuration control attribute of the Initiating Events cornerstone, and it adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, unavailability of all three chillers increased the likelihood of a loss of control air that could result in a complicated plant trip. Per Inspection Manual Chapter (IMC) 0609, Attachment 0609.04, initial screening and characterization of findings, the inspectors conducted a Phase 1 analysis and determined that this finding required a Phase 2 analysis because the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The inspector determined that the finding was of very low safety significance (Green) using the Salem plant specific Phase 2 pre-solved worksheets in accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations."

  This finding has a cross-cutting aspect in the area of human performance because PSEG personnel did not follow procedures [H.4(b)]. Specifically, revisions to the work control document for tagging the 21 chiller did not comply with the requirements of PSEG procedure SH.OP-AP.ZZ-0051, “Safety Tagging Operations.” (Section 1R13)

Cornerstone: Mitigating Systems

- Green. A self-revealing non-cited violation of Technical Specification (TS) 6.8.1.a, “Procedures and Programs,” was identified because PSEG did not adequately maintain the calibration of the Unit 2 reactor vessel level indication system (RVLIS). Specifically, scaling for both RVLIS dynamic range channels was not completed when required. This resulted in Unit 2 RVLIS being inoperable for 13-days.

  The finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and because it affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operators were not aware that both channels of RVLIS were inoperable and could have taken non-conservative actions during an inadequate core cooling or loss of coolant inventory event. Per inspection manual chapter (IMC) 0609.04, “Initial Screening and Characterization of Findings,” the inspectors conducted a Phase 1 screen and determined the finding to be of very low safety significance (Green).

  This finding has a cross-cutting aspect in the area of human performance because PSEG did not appropriately coordinate work activities as necessary to keep personnel apprised of work status and the operational impact of work activities [H.3(b)]. Specifically, PSEG did not ensure RVLIS scaling was completed per the established work control process because engineering did not adequately communicate the importance of entering the new dynamic range coefficients to the operability of the RVLIS system. (Section 1R15)

Cornerstone: Barrier Integrity

- Green. A self-revealing non-cited violation of TS 6.8.1.a, “Procedures and Programs” was identified because the 22 Containment Fan Coil Unit (CFCU) had cooling water flow to the motor cooler inadvertently isolated during a routine surveillance test. Specifically, the surveillance procedure did not include steps to operate specific gage isolation valves to place a test gage in service, and as a result technicians repositioned the wrong valves.

  This finding is more than minor because it is associated with the system, structure, and component (SSC) and barrier performance attribute of the Barrier Integrity cornerstone and it affects the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, unavailability of the 22 CFCU represented an actual loss of defense in depth of a system that controls containment pressure. Per inspection manual chapter (IMC) 0609, Attachment 0609.04, “Determining the Significance of Reactor Inspection Findings for at-power Situations,” the inspectors conducted a Phase 1 screen and determined the finding to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment isolation system and heat removal components, did not involve an actual reduction in function of hydrogen igniters in containment, and did not screen as potentially risk significant due to external initiating events.

This finding has a cross-cutting aspect in the area of human resources because PSEG did not provide complete and accurate procedures for the performance of this surveillance test [H.2(c)]. Specifically, the continuous use procedure “Service Water Fouling Monitoring Containment Fan Coil Units”, revised on May 31, 2008, did not contain procedure steps to direct the opening and closing of valves that must be manipulated to successfully perform the procedure. (Section 1R22)

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Salem Nuclear Generating Station Unit 1 (Unit 1) began the period at 93% power to support offsite transmission line maintenance. On April 1, 2008, operations returned Unit 1 to full power. On April 3, operations reduced power to 93% because a fire off site posed a risk to one of two operable transmission lines. Operations returned Unit 1 to full power on April 4. On April 20, operations reduced power to 55% because a fire off site posed a risk to one of two available transmission lines. In this instance, the transmission line was de-energized until the fire was extinguished. The line was subsequently returned to service and operations returned Unit 1 to full power on April 21. On April 26, operations reduced power to 93% because thunderstorms posed a risk to transmission grid stability. Operations returned Unit 1 to full power on April 27 and Unit 1 operated at full power for the remainder of the inspection period.

Salem Nuclear Generating Station Unit 2 (Unit 2) began the period defueled with the reactor coolant system drained to support steam generator replacement during the sixteenth refueling outage of Unit 2 (S2R16). Operations returned Unit 2 to service on May 8 and achieved 48% power before reducing power and tripping the main turbine in response to a loss of circulating water on May 9. Operations then manually tripped the reactor in response to a high steam generator water level in the 23 steam generator. Operations restarted Unit 2 on May 10 and achieved 84% power on May 12. On May 12 operations shutdown Unit 2 because PSEG determined all high steam flow protection channels were inoperable. PSEG restarted Unit 2 on May 16. Operations achieved full power on May 21 with Unit 2 power limited to 98.2% by main turbine governor valve limitations. Unit 2 remained at full power with electrical output limited by governor valve limitations for the remainder of the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 3 samples)

.1 Summer Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors completed one adverse weather inspection sample to evaluate the readiness of offsite power to the Salem units prior to the summer season when electrical grid stability can be most challenged. The inspectors verified that PSEG provided procedure requirements or guidance to monitor and maintain availability and reliability of the offsite AC power (OSP) system prior to and during adverse weather conditions. Specifically the inspectors verified that the procedures addressed:

- The actions to be taken when notified by the Electrical System Operations Center (ESOC) of the PJM interconnection that the post-trip voltage of the OSP system at Salem will not be acceptable to assure the continued operation of the safety-related loads without transferring to the emergency diesel generators.
- The compensatory actions to be performed if ESOC cannot predict the post-trip voltage.
- Re-assessment of plant risk for maintenance activities that could affect grid reliability or OSP system availability to the Salem units.
- Communication requirements between Salem and the ESOC regarding plant changes that could impact the transmission system, or the capability of the transmission system to provide adequate OSP.

The inspectors also reviewed PSEG's seasonal readiness preparations for the summer season specific to the main power transformers and the OSP system. The inspectors interviewed engineering and work control personnel and reviewed work orders and completed portions of WC-AA-107, Seasonal Readiness, Revision 7 to verify that PSEG took measures to ensure the reliability of the main power transformers and the OSP system during the summer season. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors completed one adverse weather inspection sample to evaluate Salem Generating Station's readiness and response to emergent high water levels in the Delaware River. The inspectors verified that the station took appropriate preplanned action per SC.OP-AB.ZZ-0001, Adverse Environmental Conditions. Further, the inspectors evaluated the condition of water tight doors and cover plates required by technical specifications.

b. Findings

No findings of significance were identified.

.3 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors completed one adverse weather inspection sample for the onset of hot weather. The inspectors reviewed hot weather preparations to verify PSEG adequately prepared equipment to operate reliably in extreme hot weather conditions. Specifically, the inspectors interviewed engineering and operations personnel, and walked down the service water intake structure (SWIS), the switchyard, and the switchgear & penetration area ventilation (SPAV) system. The inspectors verified that design features used to maintain these systems functional during hot weather conditions were adequately maintained. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 4 samples)

Partial Walkdown

a. Inspection Scope

The inspectors completed four partial system walk down inspection samples. The inspectors walked down the systems 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 increase plant 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 properly utilized its corrective action program to identify and resolve equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers. Documents reviewed are listed in the Attachment. The inspectors walked down the systems listed below:

- The Unit 1 and 2 control area air conditioning (CAACS) and control room emergency air conditioning (CREACS) systems following restoration to the normal alignment from the maintenance mode alignment with Unit 2 CREACS out of service on April 8;
- The Unit 2 residual heat removal (RHR) system prior to return to service following steam generator replacement during Unit 2 refueling outage S2R16 on April 17;
- The Unit 2 service water (SW) cooling for all three emergency diesel generators (EDGs) following completion of EDG maintenance during refueling outage S2R16 on April 22; and
- The Unit 1 SW system with the 12 service water pump out of service for maintenance on June 10.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05AQ - 8 samples)

.1 Quarterly Review of Fire Areas

a. Inspection Scope

The inspectors completed eight fire protection quarterly inspection samples. The inspectors performed walk downs to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles 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. The inspectors evaluated the fire protection areas listed below:

- Unit 1 and 2 service water intake structure (SWIS);
- Unit 1 and 2 outer penetration areas;
- Unit 1 and 2 auxiliary ventilation unit areas;
- Unit 2 Reactor Containment; and
- Unit 2 Reactor Plant Auxiliary Equipment Area, Elevations 45' & 55'.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A – 1 sample)

a. Inspection Scope

The inspectors reviewed performance data and interviewed the program manager responsible for implementation of NRC Generic Letter (GL) 89-13 to verify that potential heat exchanger (HX) or heat sink deficiencies were identified and that PSEG adequately resolved heat sink performance problems. Specifically, the inspectors reviewed 21 component cooling water (CCW) HX data. Inspectors evaluated trending data and verified that equipment would perform satisfactorily under design basis conditions. The method of performance monitoring was compared against NRC GL 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.

The inspectors walked down the selected components and the SW intake structure to assess the general material condition of the selected HXs and the associated SW components. The inspectors also inspected the internal components of 21 CCW HX, that was open for preventive maintenance, and observed the type and quantity of material present in the HX. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08 - 1 sample)

a. Inspection Scope

The inspectors observed a selected sample of nondestructive examination (NDE) activities in process. The inspectors also reviewed selected 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 would result in a significant increase in the risk of core damage. The observations and documentation reviews were performed to verify the activities described 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 issues identified during inservice inspection (ISI) activities. Also, the inspectors evaluated the effectiveness of the resolution of problems identified during selected ISI activities.

The inspectors reviewed PSEG's boric acid corrosion control program. The inspectors observed PSEG's boric acid walkdown inspection process inside containment at the beginning of the Unit 2 refueling outage. The inspectors verified that the walkdown inspections were thorough and that indications of boric acid leakage were recorded and 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 final resolution.

The inspectors observed the performance of one in-process NDE activity and reviewed documentation and examination reports for an additional 22 NDE activities. The inspectors reviewed seven samples of welding activities on reactor coolant system pressure boundary components and, reviewed the work packages for five repair efforts performed in accordance with the ASME Code during previous operating cycles. These observations and reviews covered ultrasonic, visual, penetrant and radiographic NDE processes.

The inspectors reviewed inspection data sheets and documentation for manual ultrasonic testing (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.

The inspectors reviewed one sample of an NDE evaluation that was initially rejected and subsequently accepted after evaluation. The inspectors also reviewed the radiographs and the examiners' interpretation of indications on several reactor coolant system welds, one main steam system weld and two feedwater system welds. These activities were performed during steam generator replacement activities.

During this refueling outage the PSEG replaced the original plant Steam Generators with new replacement steam generators provided by AREVA. The inspection of the Steam Generator Replacement activities was conducted in accordance with Inspection Procedure 50001, Steam Generator Replacement Activities. During this refueling outage, PSEG did not conduct any steam generator eddy current inspections.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 sample)

a. Inspection Scope

The inspectors completed one quarterly licensed operator requalification program inspection sample. Specifically, the inspectors observed simulator training of licensed operators on June 16, 2008. The simulated scenario involved grass loading on circulating water (CW) traveling water screens that caused a loss of two CW pumps with a third CW pump out of service for maintenance, followed by a loss of main turbine lube oil that required a turbine shutdown, followed by a failed open main steam safety valve that required operators to trip the reactor and isolate the affected main steam line. The inspectors assessed simulator fidelity and observed the crew's and the simulator instructor's critiques of operator performance. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 2 samples)

a. Inspection Scope

The inspectors completed two quarterly maintenance effectiveness inspection samples. The inspectors reviewed performance monitoring and maintenance effectiveness issues for two systems. The inspectors reviewed PSEG's process for monitoring equipment performance and assessing preventive maintenance effectiveness. The inspectors verified that systems and components were monitored in accordance with the maintenance rule program requirements. The inspectors compared documented functional failure determinations and unavailability hours to those tracked by PSEG to evaluate the effectiveness of PSEG's condition monitoring activities and to determine whether performance goals were met. The inspectors reviewed applicable work orders, corrective action notifications, and preventive maintenance tasks. The documents reviewed are listed in the Attachment. The inspectors evaluated the systems listed below:

- Unit 2 pressurizer (PZR) pilot operated relief valve (PORV) backup control air accumulator check valves, 2CA1785, 2CA1786, 2CA1787 and 2CA1788; and
- Unit 1 and 2 control room ventilation intake radiation monitors, 1R1A/B and 2R1A/B.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 7 samples)

a. Inspection Scope

The inspectors completed seven maintenance effectiveness and emergent work control inspection samples. The inspectors reviewed the maintenance activities to verify that the appropriate risk assessments were performed as specified 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. PSEG's risk management actions were reviewed during shift turnover meetings, control room tours, and plant walk downs. The inspectors also used PSEG's on-line risk monitor (Equipment Out-Of-Service workstation) to gain insights into the risk associated with these plant configurations. The inspectors reviewed notifications documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed are listed in the Attachment. The inspectors assessed the plant configurations listed below:

- Planned unavailability of the 22 component cooling water (CCW) heat exchanger following a full core offload to the spent fuel pit (SFP) during Unit 2 refueling outage S2R16 concurrent with unavailability of the 5021 offsite transmission line;
- Emergent unavailability of the 23 chiller concurrent with degradation of the 21 chiller controls and unavailability of the 5021 offsite transmission line;
- Planned unavailability of the Unit 2 control room emergency air conditioning system and unavailability of the 5021 offsite transmission line;
- Planned unavailability of the 2 and 12 station power transformers (SPT) followed by restoration of the 2 SPT and planned unavailability of the station gas turbine generator;
- Troubleshooting Unit 2 pressurizer (PZR) pilot operated relief valve (PORV) leakage that included blocking both PZR PORVs;
- Emergent unavailability of the station blackout control air compressor; and
- Unplanned unavailability of all three Unit 2 chillers.

b. Findings

Introduction: A self-revealing non-cited violation of Technical Specification (TS) 6.8.1.a, "Procedures and Programs," was identified because PSEG did not maintain adequate control of the system configuration for the Unit 2 chill water system during maintenance on the 21 chiller. Specifically, on May 27, 2008, all three Unit 2 chill water system chillers tripped due to an error in the safety tagging sequence specified by the work control documents for maintenance on the 21 chiller. This finding was determined to be of very low safety significance (Green).

Description: The Unit 2 chill water system consists of three 50% capacity chillers. The system provides cooling for the control and relay rooms and is the normal cooling supply for the emergency control air compressors. A loss of cooling to the emergency control air compressor could result in a loss of control air leading to a complicated plant trip.

During the safety tagging evolution that aligned the chill water system for the corrective maintenance on the 21 chiller, all three Unit 2 chillers tripped due to low water level in the chilled water expansion tank. This resulted in a loss of forced cooling for the relay room and the emergency control air compressor. Due to low outside air temperatures, temperatures in the relay room remained stable. The emergency control air compressor was not operating and control room cooling was provided by the Unit 1 chillers. Operators subsequently recovered chill water expansion tank level and restored the Unit 2 chill water system to service.

PSEG conducted an investigation and determined that the work control document (WCD) used for tagging the 21 chiller for maintenance included an incorrect tagging sequence. The original WCD for this maintenance properly sequenced the tagging evolution to isolate the 21 chiller prior to opening the chill water system drain valves, but before initiating the WCD, operators attempted to simplify the tagging process by grouping the service water components and the chilled water components together. When simplifying the tagging process the work center supervisor did not complete the initiation and pre-approval process as required by work control procedures. Specifically the work center supervisor did not implement the procedure step for WCD review and pre-approval that stated, “If changes are required, then return the tagging package to the initiator ...,” and “... ensure WCD changes are forwarded for field verification.” As a result the chill water drains were tagged open prior to closing and tagging the chill water outlet valve on the 21 chiller. This resulted in draining the system and a low water level in the chill water expansion tank that caused the loss of all three Unit 2 chillers.

PSEG’s corrective actions included training for the operators involved in this event. The training reinforced the procedural requirement for independent checks of all pre-approved work control documents that are modified in any manner.

<u>Analysis</u>: PSEG did not perform independent checks of the WCD following completion of revisions to the tagging sequence for the 21 chiller. The inspectors determined that this was a performance deficiency because procedure SH.OP-AP.ZZ-0051, “Safety Tagging Operations,” required that the revised WCD be returned to the initiator for review and forwarded for field verification, but these steps were not completed. This performance deficiency is more than minor because it is associated with the configuration control attribute of the Initiating Events cornerstone, and it adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, unavailability of all three chillers increased the likelihood of a loss of control air that could result in a complicated plant trip. Per Inspection Manual Chapter (IMC) 0609, Attachment 0609.04, initial screening and characterization of findings, the inspectors conducted a Phase 1 analysis and determined that this finding required a Phase 2 analysis because the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available.

The inspectors determined that the finding was of very low safety significance (Green) using the Salem plant specific Phase 2 pre-solved worksheets in accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” The estimated increase in core damage frequency was approximately 1 in 100,000,000 (actual was E-8) assuming the reactor was operating at full power and all three chillers were out of service for two hours which resulted in an increase in the loss of control air (LCA) initiating event frequency. An evaluation of the Salem plant specific Phase 2 pre-solved worksheets showed that two sequences were dominant. The first involved a LCA with a failure of the following components/functions: Station blackout compressor, auxiliary feedwater, and high pressure recirculation. The second involved a LCA with the failure of the following components/functions: Station blackout compressor, auxiliary feedwater, and feed and bleed. Service water has the ability to be manually aligned to the Unit 2 control air compressor through a spool piece and was available throughout the event and could have provided some mitigation capability.

This finding has a cross-cutting aspect in the area of human performance because PSEG personnel did not follow procedures [H.4(b)]. Specifically, revisions to the work control document for tagging the 21 chiller did not comply with the requirements of PSEG procedure SH.OP-AP.ZZ-0051, "Safety Tagging Operations."

Enforcement: Technical Specification 6.8.1.a requires the establishment, implementation and maintenance of written procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 1 recommends administrative activities, including equipment control (e.g., locking and tagging) be covered by written procedures. PSEG procedure SH.OP-AP.ZZ-0051, "Safety Tagging Operations," requires that when revising an approved tagging package, the initiator, the qualified operator, and the approving supervisor each review blocking points for adequacy. Contrary to the above, when revising an approved work control document to hang tags in preparation for performing maintenance on the 21 chiller, work control personnel did not ensure the required reviews and approvals were completed. As a result the chill water drains were tagged open prior to closing the chill water outlet valve on the 21 chiller. This caused the system to drain and a low water level in the chill water expansion. When the chill water expansion tank drained the two operating chillers tripped and there were no operable chillers on Unit 2 for a period of two hours on May 27, 2008. Because this finding is of very low safety significance and has been entered into the corrective action program in notification 20371577, this violation is being treated as a NCV, consistent with section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2008003-01, Salem Unit 2 Loss of All Three Chillers)**

1R15 Operability Evaluations (71111.15 - 8 samples)

a. Inspection Scope

The inspectors completed eight operability evaluation inspection samples. The inspectors reviewed the operability determinations for degraded or non-conforming conditions associated with:

- Restoration of the Unit 1 and 2 control room envelope (CRE) to operability following emergent failure of radiation monitor 1R1B-2;
- 13 containment fan coil unit (CFCU) following unsatisfactory computed biofouling margin in the 13 CFCU motor cooler;

- Unit 2 service water system following incorrect installation of WEKO seal backing rings throughout the system during S2R16;
- Unit 2 reactor vessel level indication system (RVLIS) following a refueling outage that included steam generator replacement;
- Unit 1 reactor protection system (RPS) given a 6% mismatch between indicated steam flow and feedwater flow;
- Unit 2 RPS given a mismatch of up to 14% between indicated steam flow and feedwater flow;
- The 13 component cooling water (CCW) pump with an oil leak from the outboard pump bearing; and
- The 21 and 22 safety injection (SJ) pumps with high vibrations when the pumps were operated at full flow during technical specification surveillance testing.

The inspectors reviewed the technical adequacy of the operability determinations to ensure 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. Documents reviewed are listed in the Attachment.

a. <u>Findings</u>

<u>Introduction</u>: A self-revealing non-cited violation of Technical Specification (TS) 6.8.1.a, "Procedures and Programs," was identified because PSEG did not adequately maintain the calibration of the Unit 2 reactor vessel level indication system (RVLIS). Specifically, scaling for both RVLIS dynamic range channels was not completed when required. This resulted in Unit 2 RVLIS being inoperable for 13-days. This finding was determined to be of very low safety significance (Green).

<u>Description</u>: The RVLIS measures reactor vessel water level when reactor coolant pumps are off and relative reactor coolant void content, dynamic range indication, when reactor coolant pumps are running. Inaccurate RVLIS indication could cause operators to take non-conservative actions during an inadequate core cooling or loss of coolant inventory event. The purpose of the RVLIS scaling procedure is to verify proper calibration of the RVLIS dynamic range indication. The procedure is performed on an 18-month frequency following each refueling outage. The scaling procedure records operating data for the plant during heat up after each refueling outage. The data is used to calculate dynamic head curve coefficients that are then used by RVLIS to determine dynamic range indication.

On May 20, 2008, both channels of RVLIS dynamic range indication began to indicate high off scale as Unit 2 approached full power after its refueling outage. As a result, operators declared both channels of RVLIS dynamic range inoperable per Technical Specification 3.3.3.7 and implemented compensatory measures.

PSEG conducted an apparent cause evaluation and determined that data obtained during plant heat up indicated that new dynamic head curve coefficients needed to be entered into the system to ensure accurate RVLIS indication. Maintenance personnel did not complete RVLIS dynamic range scaling in accordance with SC.IC-PT.RVL-0001, "RVLIS Level Output Scaling Adjustments and Heat-up Data Collection," before a maintenance supervisor inappropriately closed the associated work order on May 7. Technicians had recorded the required plant operating data and were provided the new dynamic range head curve coefficients before May 8, but did not enter the new coefficients into RVLIS until after the RVLIS dynamic range indicated high off scale on May 20. Inadequate work coordination between Operations, Maintenance, and Engineering caused plant operators to believe that RVLIS was operable even after an engineer discovered that the new dynamic head coefficients had not been entered. On May 8 the engineer, who knew the coefficients were required to maintain RVLIS indication accurate, consulted with the responsible maintenance supervisor and initiated a new work order to install the coefficients. However, because the knowledgeable engineer did not emphasize the importance of entering the new coefficients through discussions and because the work order did not accurately describe the impact on operability, the operators did not recognize that this work was important to RVLIS operability. The operators denied permission for a maintenance technician to install the new coefficients under the new work order because the control room was busy with plant startup activities. As a result, the new coefficients were not installed until May 20 after operators discovered that RVLIS was inoperable because dynamic range indication began to indicate off scale.

PSEG entered this issue into the CAP and implemented several corrective actions that included a review of the status of other safety-related work completed by the applicable maintenance supervisor to verify accurate work documentation; procedure enhancements to both maintenance and operations procedures; and personnel actions to enforce standards, improve teamwork and enhance communications through shift turnover.

Analysis: PSEG did not complete RVLIS scaling in accordance with SC.IC-PT.RVL-0001, "RVLIS Level Output Scaling Adjustments and Heat-up Data Collection." The inspectors determined that this was a performance deficiency because the original work order and the RVLIS scaling procedure directed that the procedure, including entry of the new coefficients, be completed. As a result, operators were not aware that both channels of RVLIS were inoperable for more than 13-days. The performance deficiency is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and because it affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operators were not aware that both channels of RVLIS were inoperable and could have taken non-conservative actions during an inadequate core cooling or loss of coolant inventory event. Per inspection manual chapter (IMC) 0609.04, "Initial Screening and Characterization of Findings," the inspectors conducted a Phase 1 screen and determined the finding to be of very low safety significance (Green) because the performance deficiency was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events.

This performance deficiency has a cross-cutting aspect in the area of human performance because PSEG did not appropriately coordinate work activities as necessary to keep personnel apprised of work status and the operational impact of work activities. Specifically, PSEG did not ensure RVLIS scaling was completed per the established work control process because engineering did not adequately communicate the importance of entering the new dynamic range coefficients to support operability of the RVLIS system. [H.3(b)]

Enforcement: Technical Specification 6.8.1.a requires establishment, implementation and maintenance of written procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 9 requires that maintenance affecting the performance of safety-related equipment be performed per written procedures, documented instructions or drawings appropriate to the circumstances. Contrary to the above, on May 7, 2008, PSEG collected required data, obtained new dynamic range head curve coefficients, but did not complete the RVLIS scaling procedure by entering the coefficients into the RVLIS system. Consequently, accident monitoring instrumentation required by technical specifications was inoperable for approximately 13-days. Because this finding is of very low safety significance and has been entered into the corrective action program in notification 20371034, this violation is being treated as a NCV, consistent with section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2008003-02, Salem Unit 2 Loss of Reactor Vessel Level Indication System)**

1R18 Plant Modifications (71111.18 – 2 samples)

.1 Temporary Modification

a. Inspection Scope

The inspectors completed one plant modification inspection sample. The inspectors reviewed Temporary Configuration Change Package (TCCP) -08-002 for the 15 service water pump packing arrangement. This configuration change altered the packing arrangement in the stuffing box due to wear on the installed shaft sleeve. The inspectors review included system walk downs and comparison of the temporarily modified configuration to the UFSAR description. Inspectors reviewed the 10 CFR 50.59 screening against the system design basis documentation, and verified that the modifications did not affect system operability.

b. Findings

No findings of significance were identified.

.2 Permanent Modification

a. Inspection Scope

The inspectors completed one plant modification inspection sample. The inspectors reviewed permanent modifications to the Salem Unit 2 Containment Fan Coil Unit (CFCU) service water flow control under change number DCP 80089131. This review included system walk downs, interviews with plant engineers, and functional comparison of the new control scheme to the UFSAR description. The inspectors also reviewed design adequacy of the modification, preparation, staging, and implementation of the modification, and post modification testing.

This modification replaced a system that varied service water flow for normal and accident conditions with a simplified scheme that provided a fixed resistance service water flow to the CFCUs under normal and accident conditions. This modification simplified the SW system and eliminated numerous components and instruments that had active safety functions. This modification was made following approval of License Change Request S06-10. Each of the five CFCUs located in the Unit 2 containment were included in this permanent modification. This modification was performed during the Unit 2 refuel outage (2R16).

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 7 samples)

a. Inspection Scope

The inspectors completed seven post-maintenance testing inspection samples. The inspectors observed portions of and/or reviewed the results of the post-maintenance test activities. The inspectors verified that the effect of testing on the plant was adequately addressed by control room and engineering personnel; testing was adequate for the maintenance performed; acceptance criteria were clear, demonstrated operational readiness and were consistent with design and licensing basis documentation; test instrumentation was calibrated, and the appropriate range and accuracy for the application; tests were performed, as written, with applicable prerequisites satisfied; and equipment was returned to an operational status and ready to perform its safety function. Documents reviewed are listed in the Attachment. The inspectors evaluated the post-maintenance tests for the maintenance items listed below:

- Work order (WO) CM980824163, 2A emergency diesel generator (EDG) K1C relay replacement;
- WO 60054063, weld repairs to the 22 component cooling heat exchanger (CCHX) north cover liner;
- WO 60076709, replacement of 2N43 power range neutron detector;
- WO 60076388 and 60076389, adjustment of 21SJ44 and 22SJ44 stroke times and valve leakage troubleshooting;
- WO 80083522, main steam flow protection and indication following replacement of Unit 2 steam generators and high pressure turbine;
- WO 30145027, reactor vessel level instrumentation system (RVLIS) return to service following replacement of Unit 2 steam generators; and
- WO 30126880, 1A1 28 volt DC battery charger following 18 month preventive maintenance including replacement of selected charger circuit cards and filter capacitors.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

Unit 2 Refueling Outage. The inspectors observed and/or reviewed the following refueling outage activities to verify that operability requirements were met and that risk, industry experience, and previous site specific problems were considered. Documents reviewed for this inspection are listed in the Attachment.

The inspectors reviewed the schedule and risk assessment documents associated with the 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 verified that PSEG managed the outage risk commensurate with the outage plan.

The inspectors observed portions of the shutdown and cool down processes and monitored PSEG controls over the outage activities. The inspectors also verified that cool down rates were within technical specification limitations.

The inspectors observed conditions within containment for indications of unidentified leakage and damaged equipment.

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 four Unit 2 steam generators were replaced during this refuel outage. The old steam generators were removed and new steam generators were installed through the containment equipment hatch. Inspection of these specific activities is discussed in section 4OA5.

The inspectors verified that tagged equipment was properly controlled and equipment configured to safely support maintenance work. Specifically, tags hung in containment to support the steam generator replacement were checked for accuracy of the tag versus the label, as well as the valve position specified.

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 reactor coolant system (RCS) level and temperature and that indications were within the expected range for the operating mode.

The inspectors verified that offsite and onsite electrical power sources were maintained in accordance with TS requirements and consistent with the outage risk assessment. Periodic walk downs 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.

The inspectors observed the Unit 2 RCS draining to the mid-loop condition on April 27, 2008. RCS 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.

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 that could block containment sumps was removed. Portions of mode changes and reactor startup were observed and reviewed for compliance with applicable procedures and TS.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 7 samples)

a. Inspection Scope

The inspectors completed seven surveillance testing inspection samples. The inspectors observed portions of and/or reviewed results for the 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 TS requirements, the UFSAR, and ASME Section XI for pump and valve testing. Documents reviewed are listed in the Attachment. The inspectors evaluated the surveillance tests listed below:

- SC.MD-EU.CRN-0013, "Fuel Handling Crane Daily Use Inspection," during Unit 2 core reload activities;
- S2.OP-ST.SJ-0021, "RCS Pressure Isolation Leakage Test 2RH1 and 2RH2 Valves (Containment Isolation Valves (CIV))";
- S2.OP-ST.MS-0003, Steam Line Isolation and Response Time Testing;
- S2.OP-ST.AF-0003, Inservice Testing – 23 Auxiliary Feedwater Pump (IST);
- S2.OP-ST.AF-0006, Inservice Testing – Auxiliary Feed Water Valves;
- S2.OP-ST.AF-0009, Plant Systems - Auxiliary Feedwater; and
- S2.OP-ST.SJ-0020, Periodic Leakage Test, RCS Pressure Isolation Valves (CIV).

b. Findings

Introduction: A self-revealing non-cited violation of Technical Specification (TS) 6.8.1.a, "Procedures and Programs," was identified because the 22 Containment Fan Coil Unit (CFCU) had cooling water flow to the motor cooler inadvertently isolated during a routine surveillance test. Specifically, the surveillance procedure did not include steps to operate specific gage isolation valves to place a gage in service, and as a result technicians repositioned the wrong valves. This finding was determined to be of very low safety significance (Green).

Description: The Unit 2 containment cooling system consists of five CFCUs designed to maintain containment temperature below design temperature limits during normal operation, and maintain containment pressure below design pressure limits during accident conditions by condensing steam vapor in the containment.

During the performance of a surveillance test on the 22 CFCU, technicians measured zero service water flow through the 22 CFCU motor cooler. The technicians immediately notified the control room and operators declared the 22 CFCU inoperable and subsequently removed it from service.

PSEG conducted an investigation of the event and determined that one of two technicians assigned to perform the surveillance testing had inadvertently closed the 22 CFCU motor cooler service water inlet and outlet valves while attempting to place a differential pressure gage, installed to collect test data, in service. Both of these valves were normally locked open, but the technician was able to reposition them without removing the locks. PSEG determined through its review that the continuous use procedure "Service Water Fouling Monitoring Containment Fan Coil Units" that was used to complete the testing did not contain specific steps that identified the valves that the technicians needed to open to place the differential pressure gage in service. Due to the lack of information provided in the procedure, the technician relied on his knowledge and training for the system to identify those valves. As a result the technician incorrectly shut the locked open motor cooler service water inlet and outlet isolation valves, which rendered the 22 CFCU inoperable.

Corrective actions taken by PSEG were to immediately restore the cooler isolation valves to their correct positions and to revise the surveillance procedure to add individual signoff steps for each valve manipulation and action taken. In addition, because the valves that were repositioned were locked valves, PSEG also performed a check of all locked valves on Unit 2 to verify compliance with the requirements of the locked-valve program.

Analysis: The procedure used to perform surveillance testing on the 22 CFCU did not contain procedure steps that directed technicians to open valves to place a differential pressure gage in service. As a result a technician inadvertently closed two normally locked open valves that isolated cooling water flow to the CFCU. The inspectors determined that this was a performance deficiency because TS 6.8.1.a requires the establishment, implementation and maintenance of adequate written procedures for operations discussed in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978. The issue was within PSEG's ability to foresee and correct because PSEG recently revised the procedure, but did not ensure that it conformed to continuous use procedure standards. This finding is more than minor because it is associated with the system, structure, and component (SSC) and barrier performance attribute of the Barrier Integrity cornerstone and it affects the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, unavailability of the 22 CFCU represented an actual loss of defense in depth of a system that controls containment pressure. Per inspection manual chapter (IMC) 0609, Attachment 0609.04, "Determining the Significance of Reactor Inspection Findings for at-power Situations," the inspectors conducted a Phase 1 screen and determined the finding to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment isolation system and heat removal components, did not involve an actual reduction in function of hydrogen igniters in containment, and did not screen as potentially risk significant due to external initiating events.

This finding has a cross-cutting aspect in the area of human resources because PSEG did not provide complete and accurate procedures for the performance of this surveillance test [H.2(c)]. Specifically, the continuous use procedure "Service Water Fouling Monitoring Containment Fan Coil Units", revised on May 31, 2008, did not contain procedure steps to direct the opening and closing of valves that must be manipulated to successfully perform the procedure.

Enforcement: Technical Specification 6.8.1.a requires the establishment, implementation and maintenance of written procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 8 recommends that specific procedures be written for surveillance tests, inspections and calibrations. Contrary to the above, PSEG procedure S2.OP-PT.SW-0007, "Service Water Fouling Monitoring Containment Fan Coil Units," was inadequate for performing surveillance testing on the 22 CFCU. Cooling water flow to the 22 CFCU motor cooler was inadvertently isolated, resulting in one hour of unavailability of the 22

CFCU on June 4, 2008. Because this finding is of very low safety significance and has been entered into the corrective action program in notification 20372695, this violation is being treated as a NCV, consistent with section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2008002-03, Salem Unit 2 22 CFCU Valves Mispositioned)**

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 8 samples)

a. Inspection Scope

Based on PSEG's schedule of work activities during the Unit 2 refueling outage (S2R16), the inspectors selected three jobs (reactor disassembly, reactor coolant pump motor replacement, in-service inspection) performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation. The inspectors observed work that was estimated to result in the highest collective doses, involved diving activities in or around spent fuel or highly activated material, or that involved potentially changing (deteriorating) radiological conditions. The inspectors reviewed all radiological job requirements (radiation work permit requirements and work procedure requirements). The inspectors observed job performance with respect to these requirements and verified 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.

During job performance observations, the inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors verified that they were aware of the significant radiological conditions in their workplace, the radiation work permit 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 all radiation protection work requirements. The inspectors verified that they were aware of the radiological conditions in their workplace and the radiation work permit controls/limits, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors identified exposure significant work areas within radiation areas, high radiation areas (<1 R/hr), or airborne radioactivity areas in the plant and review associated PSEG controls and surveys of these areas to determine if controls (e.g., surveys, postings, barricades) were acceptable. The areas reviewed by the inspectors included the containment, auxiliary building and spent fuel building. With a survey instrument, the inspectors walked down these areas or their perimeters to determine: whether prescribed radiation work permits, procedure, and engineering controls were in place, whether PSEG surveys and postings were complete and accurate, and whether air samplers were properly located.

The inspectors reviewed radiation work permits used to access these and other high radiation areas and identify what work control instructions or control barriers were specified. The inspectors used plant-specific TS high radiation area requirements as the standard for the necessary barriers. 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 verified that workers know what actions are required when their electronic personal dosimeter noticeably malfunctions or alarms.

The inspector evaluated PSEG performance against the requirements contained in 10 CFR 20, and Unit 2 Technical Specification 6.12.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 4 samples)

a. Inspection Scope

The inspectors obtained from PSEG a list of work activities ranked by actual/estimated exposure that were in progress during the current outage (2R16) and selected the three work activities of the highest exposure significance (see section 2OS1 above).

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors verified that PSEG had established procedures, and engineering and work controls that were based on sound radiation protection principles, to achieve occupational exposures that were ALARA. The inspectors verified that PSEG had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in PSEG's ALARA planning for these work activities. The inspectors also reviewed the reason for any inconsistencies between intended and actual work activity doses.

Based on scheduled work activities and associated exposure estimates, the inspectors selected three work activities in radiation areas, airborne radioactivity areas, or high radiation areas for observation. The inspectors concentrated on work activities that presented the greatest radiological risk to workers. The inspectors evaluated PSEG's use of ALARA controls for these work activities by evaluating PSEG's use of engineering controls to achieve dose reductions.

The inspectors evaluated PSEG performance against the requirements contained in 10 CFR 20.1101.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 1 sample)

a. Inspection Scope

The inspectors identified the types of portable radiation detection instrumentation used for job coverage for 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 committed effective dose equivalent.

The inspectors evaluated PSEG performance in this area against the requirements contained in 10 CFR 20.1501, 10 CFR 20.1703 and 10 CFR 20.1704.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151 - 6 samples)

a. Inspection Scope

The inspectors reviewed PSEG submittals for the Unit 1 and Unit 2 initiating events cornerstone performance indicators discussed below. To verify the accuracy of the PI data reported during this period the data was compared to the PI definition and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 5.

Cornerstone: Initiating Events

- Unit 1 and Unit 2 unplanned scrams
- Unit 1 and Unit 2 unplanned scrams with complications
- Unit 1 and Unit 2 unplanned power changes

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 2 samples)

.1 Review of Items Entered into the Corrective Action Program (CAP):

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 all 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.

b. Assessment and Observations

No findings of significance were identified.

.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 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 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 December 1, 2007, through May 31, 2008, 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. 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 equipment reliability issues. Several unplanned equipment outages were required to make repairs to safety-related chillers, containment fan coil units, and service water pump strainers. In addition, the 13A circulating water screen was unreliable during the river grassing season this spring. The inspectors determined PSEG is aware of these areas identified through this trend review and is appropriately addressing these issues.

.3 Annual Sample: Human Performance, Procedure Use and Adherence

a. Inspection Scope

The inspectors reviewed the actions PSEG had taken to improve procedure use and adherence at the station. This sample evaluated PSEG's scope of efforts and progress in the area of procedure compliance for the period of January 2008 through June 2008.

b. Findings and Observations

No findings of significance were identified.

On August 31, 2007, the NRC identified a substantive cross-cutting issue in the area of human performance with a cross-cutting theme in the aspect of procedure compliance. This substantive cross-cutting issue was maintained in the end of cycle assessment letter because PSEG did not meet the criteria for clearing the substantive cross-cutting issue. The root cause evaluation was completed by PSEG in December 2007, and some of the corrective actions were still in progress at the end of the assessment period.

The cause evaluation confirmed that personnel did not consistently meet management procedure use and adherence expectations. PSEG determined that the root cause was inadequate reinforcement and oversight of procedure use and adherence expectations. As a result of inadequate reinforcement and oversight, the organization tolerated substandard procedures and a disregard for procedural steps perceived to be of low value.

The six month period following the completion of the root cause evaluation, January 2008 through June 2008, included approximately two months of outage preparation and two months of outage execution for steam generator replacement on Unit 2. A review of procedure compliance during the outage preparation period revealed only minor issues. These issues were documented in PSEG's corrective action process or the PSEG fundamentals management system (FMS). Procedure compliance during the refuel outage and steam generator replacement on Unit 2 was also satisfactory. A large volume of clearance activities, maintenance activities, and testing activities were performed with only minor issues being noted.

During the week of June 16, 2008, PSEG conducted a self-assessment on procedure use and adherence. The self-assessment concluded that workforce behavior related to procedure use and adherence has improved and is in compliance with the standard, HU-AA-104-101, Procedure Use and Adherence. Several areas were observed and evaluated to make this determination. Results from 13 of 14 field observations were that the work force is adhering to procedures and work instructions, using place keeping, stopping when encountering ambiguity, and engaging supervisors in resolving issues. Interviews with about 40 maintenance workers, security officers, engineers, operators, chemists, and RP technicians confirmed that procedure use and adherence standards are well known and reinforced by station management personnel. FMS and manager in the field reports reviewed indicated that procedure use and adherence is receiving priority attention from supervisors, superintendents, and managers in the field.

Inspectors independently checked for procedure compliance issues while work is in progress, and found that written procedures are being followed as technicians perform plant operations and maintenance. Technicians performing work were willing to discuss their compliance with procedures, and notifications were written to document changes requested to procedures to improve procedure quality.

Dynamic learning activities in procedure use and adherence were completed by maintenance, engineering, and operations departments. Chemistry had led the effort and completed their dynamic learning activity (DLA) training in 2007. These activities were one to four hours in duration, and were completed by each person prior to June 30, 2008. The DLAs are training exercises, where each technician is trained and provided feedback. In addition, maintenance has scheduled an out of the box evaluation for each technician during the third quarter of 2008. These evaluations will continue to reinforce the expectations for procedure use at the job site. All corrective actions from the root cause evaluation have been completed.

One area of the self-assessment that did not meet expectations was in the area of procedure quality. Procedure revision backlogs have not been reduced to target numbers, and a recovery plan is included in the self-assessment. Reduction of this backlog to reach year end goals is important to the station's continued improvement in procedure use and adherence.

4OA3 Event Followup (71153 - 2 samples)

.1 Unit 2 Manual Reactor Trip

a. Inspection Scope

The inspectors responded to a manual reactor trip that occurred on May 9, 2008. As reactor power was being reduced in response to a partial loss of circulating water, the level in 23 steam generator approached the hi-hi level setpoint. Plant operators manually tripped the reactor due to the high steam generator level. The inspectors observed control room operators establish and maintain stable hot-standby conditions. The inspectors walked down all control board indications for abnormalities, and interviewed operators for additional insights on equipment performance.

The inspectors discussed the reactor trip with PSEG's investigation team, managers, and engineers. The inspectors reviewed the initial investigation report and post-reactor trip report, and observed the plant operations review committee meeting on restart issues.

PSEG's initial investigation determined that the partial loss of circulating water was caused by a loss of power to the Unit 2 circulating water traveling screens. As debris built up on the traveling screens, differential level increased over time to reach the circulating water pump trip set point. The main turbine was tripped in accordance with the circulating water abnormal operating procedure, and the level controller for the 23 steam generator sensed an error that caused the controller to switch to manual. Steam generator level increased, and the plant operator tripped the reactor. Corrective actions resulting from this event were to repair the faulted electrical circuit powering the circulating water screens, and adjust the error sensing zero signal for the Unit 2 steam generators.

b. Findings

No findings of significance were identified.

.2 Unit 2 Forced Shutdown

a. Inspection Scope

The inspectors responded to a reactor shutdown that occurred on May 12, 2008. During power ascension following a refuel outage and steam generator replacement, plant operators identified a significant mismatch between feed flow indication and steam flow indication for each of the four steam generators. PSEG determined that the steam flow indication and input to reactor protection was calibrated improperly, and declared all eight steam flow protection circuits inoperable. The reactor was shut down in accordance with TS 3.0.3. The inspectors observed control room operators establish and maintain stable hot standby conditions.

The inspectors discussed the miscalibration and plant shutdown with PSEG's investigation team, managers, and engineers. The inspectors reviewed the initial investigation report, and observed the plant operations review committee on restart issues.

PSEG's initial investigation determined that the steam flow calibration errors were due to multiple errors with the data entered into the plant computer scaling model. These errors were corrected prior to plant restart, and steam flow indication was carefully monitored during plant power ascension.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Steam Generator Replacement Inspection Procedure 50001

a. Inspection Scope

The inspectors verified that engineering evaluations and design changes associated with steam generator replacement (SGR) were completed in conformance with requirements in the facility license, the applicable codes and standards, licensing commitments, and the regulations.

The inspectors reviewed the applicable Design Change Packages (DCP's) that controlled the replacement of the steam generators. All DCP's were reviewed in accordance with the requirements of IP 71111.02, Evaluation of Changes, Tests and Experiments and IP 71111.17B, Permanent Plant Modifications.

The inspectors reviewed the administrative procedures that were used to control the screening, preparation, and issuance of the associated safety evaluations to ensure that the procedure adequately covered the requirements of regulation 10 CFR 50.59.

The inspectors reviewed the DCP's and implementing Work Packages for the site haul path for movement of the replacement steam generators (RSGs) and the old steam generators (OSGs). The review included the engineering evaluation of potentially affected underground piping systems, structures and electrical conduits for the operating unit (Unit 1) and for the necessary support equipment for the shutdown operation of Unit 2. The inspectors also reviewed the rigging and handling procedures for the OSGs and the RSGs both inside and outside of containment. Access to containment for this replacement project was through the equipment hatch. No modification to the permanent containment structure was necessary for the steam generator replacement.

The inspectors performed a review of applicable procedures, work practices, and documentation of the steam generator replacement project (SGRP) to assess the control of safety related aspects associated with the major phases of the SG replacement. This inspection was focused on SGRP activities that restore the pressure boundary of the reactor coolant system (RCS), secondary systems, and affected containment systems, exclusion of foreign materials and plant modifications that could affect plant risk during the replacement activities or subsequent plant operation. The inspectors also examined the affected steam generator supports and affected piping system supports.

The inspectors performed a review of those procedures that governed field activities involving the cutting and machining of weld preparations at existing main steam (MS), feedwater (FW) and reactor coolant system (RCS) connections on the old steam generators (OSG). The inspectors examined welding procedures, welder qualifications, weld filler metal selection, non-destructive test procedures, examiner qualifications, acceptance criteria and test results for compliance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Sections IX and XI. The inspectors examined training and qualification records for selected personnel performing welding and non-destructive examination. The inspectors selected a sample of Work Packages (WP) for review. The selection of these field planning and work documents was based on their risk significance and represented the installation by welding of FW, MS and RCS piping, fittings and pipe supports. The tracking and processing of the work planning and control documents was examined. The work control documents selected were found to be appropriate for the work tasks specified and contained provisions and 'hold points' to monitor weld progress, provide necessary in-process NDE and to close out the activities at the completion of work.

The inspectors reviewed the manufacturing records and primary and secondary shop fabrication hydrostatic testing records. The inspectors reviewed the non-destructive test results of the pre-service examinations of the RSG shell, nozzle and attachment welds and the baseline examination of the steam generator tubes. The inspectors also reviewed the SG degradation assessment developed in advance of the pre-service tube inspections (PSI) to be performed with respect to the limited range of degradation mechanisms that may pertain to the as-built condition.

Prior to beginning replacement activities the inspectors reviewed the rigging, lifting and handling procedures used to move the old steam generators from containment and move the new steam generators into place inside containment. The inspectors verified that PSEG addressed the potential impact on system and equipment from Unit 1, the operating unit, and had analyzed and addressed the potential for damage to existing plant SSCs due to the moving of these heavy loads. The inspectors directly observed the lifting, rigging and removal of the OSGs No. 22 and No. 24 out of containment

The inspectors verified that PSEG provided temporary power to construction efforts inside the Unit 2 containment and that these power sources were independent from the operating Unit 1 and Unit 2 shutdown safety significant electrical loads.

The inspectors also reviewed the security considerations associated with vital and protected area barriers that may have been affected by the steam generator replacement activities. The inspectors also conducted a walkdown and review of security boundaries affected by the steam generator replacement project and conducted a review of PSEG's compensatory measures.

Specific work activities inspected included observation of the qualification of welders using the automated process for the welding of the RCS piping, MS piping and FW piping. Also, the inspectors observed the machining of weld preparations on selected samples of MS, FW and RCS piping spools and assessed the dimensional accuracy for comparison with specification requirements. The inspectors reviewed SGT work package No. 3065C, RCS Machining and Welding SG-23, and discussed the RCS hot and cold leg piping end preparation machining tasks with the machinist performing the activity. The inspectors observed the shop welding of various piping spools in the pipe fabrication shop prior to their movement to the field for installation. The inspectors observed the field fit-up, tack welding and final welding of a sample of these components to their respective steam generator nozzles.

One field weld from each of the MS, FW and RCS piping was selected by the inspectors for an in depth review of fabrication work documentation, including the severance at the steam generator nozzle to the completion of re-attachment to the steam generator including in process and final non-destructive testing of the new welds. The final non-destructive tests for the acceptance of these risk significant welds involved surface and radiographic testing. Also, ultrasonic testing was required for each weld to establish baseline quality levels for comparison during subsequent ASME Section XI exams (in-service inspection). The inspectors reviewed the test activities and inspection reports to assess the completeness of the examination processes. Also, the inspectors conducted interviews with design and field engineers, and with construction and examination personnel responsible for implementing these activities.

The inspectors reviewed the post-steam generator replacement NDE of all new welds for the risk significant RCS, MS and FW systems. The inspectors also reviewed the in-service ultrasonic inspection baseline examinations performed on the new RCS, MS and FW system piping welds.

The inspectors reviewed the SG post-installation test program and PSEG documentation for the RCS system pressurization leak test and walkdown during initial plant startup. The inspectors also reviewed the testing documentation used to verify that RCS flow for the new steam generators conformed to design predictions and a sample of calibration and testing procedure results for plant instrumentation affected by the steam generator replacement.

The inspectors reviewed the planned startup testing procedure for the replacement steam generators. The testing procedure was part of the OSG replacement project plan and was evaluated as part of the steam generator replacement 10 CFR 50.59 evaluation.

The inspectors reviewed the following radiation protection program controls, planning, and preparation, against the requirements contained in 10 CFR 20, 10 CFR 20.1101, and Unit 2 Technical Specification 6.12.:

- <u>ALARA planning</u>. The inspectors reviewed ALARA planning documents, including ALARA plans and micro-ALARA plans written in support of the steam generator replacement project, and also examined in-progress work reviews. Eleven ALARA plans and one micro-ALARA plan were written in support of the project. The inspectors verified that the ALARA planning process took into account both individual and collective dose considerations.

  The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined that PSEG had established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that were ALARA. The inspectors determined that PSEG had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

- <u>Dose estimates and dose tracking</u>. The inspectors reviewed pre-outage estimates and goals, together with in-progress dose tracking data. The inspectors verified that daily dose tracking data was provided to management, and that individual and work group data was also readily available.

  The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities. The inspectors determined the causes for any inconsistencies between intended and actual work activity doses. Total dose for the steam generator replacement project was 15 person-rem below the pre-outage estimate.

- <u>Exposure controls including temporary shielding</u>. The inspectors verified that the ALARA plans included the use of temporary shielding and other engineering controls to minimize exposure. The inspectors conducted numerous in-field observations to verify that shielding packages were installed as described, and that engineering controls were being appropriately utilized.

- Contamination controls. The inspectors conducted numerous observations of contamination controls implemented in the radiologically controlled area, especially in the containment building. The inspectors verified that contamination controls were appropriately implemented, that barriers were appropriately installed and that sufficient radiological postings were available to inform the workers.

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

- Radiological material management. The inspectors verified that materials removed from the old steam generators and other contaminated materials were being appropriately identified, segregated and processed. The inspectors made their determination via direct observations and review of PSEG documents.

- Radiological work plans and controls. The inspectors reviewed PSEG procedures, radiation work permits and ALARA plans related to the steam generator replacement. The inspectors conducted direct observations of PSEG work in progress to verify that the plans and controls described were being appropriately implemented.

  The inspectors observed work that was estimated to result in the highest collective doses, involved diving activities in or around spent fuel or highly activated material, or that involved potentially changing (deteriorating) radiological conditions. The inspectors reviewed the radiological job requirements (radiation work permit requirements and work procedure requirements). The inspectors observed job performance with respect to these requirements. The inspectors determined if 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.

  The inspectors reviewed radiation work permits used to access these and other high radiation areas and identify what work control instructions or control barriers had been specified. The inspectors used plant-specific technical specification high radiation area requirements as the standard for the necessary barriers. 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 verified that workers know what actions are required when their electronic personal dosimeter noticeably malfunctions or alarms.

- Emergency contingencies. The inspectors reviewed ALARA plans and radiation work permits utilized for the steam generator replacement, and verified that back-out dose rates and other radiological limits were identified and communicated to the workers. The inspectors did not identify any instances were radiological limits were exceeded.

- Project staffing and training plans. The inspectors reviewed PSEG's radiation protection staffing plan, and verified that adequate radiation protection technicians and other radiological staff were available on both day and night shifts.

  During job performance observations, the inspectors observed radiation protection technician performance with respect to all radiation protection work requirements. The inspectors determined if they were aware of the radiological conditions in their workplace and the radiation work permit controls/limits, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

- Radiological safety plans for storage of retired steam generators and components. The inspectors reviewed PSEG's plan for temporary storage of the old steam generators at the interim low level radioactive waste storage facility. The inspectors toured the storage location, and verified that dose rate limits set forth in PSEG's plan were not being exceeded. The inspectors reviewed PSEG's transport and disposal plan for the OSGs, scheduled to take place in late 2008.

b. Findings

During the sixteenth refueling outage at Unit 2, all four steam generators and the high pressure turbine were replaced. These replacements resulted in changes to various plant parameters, including main steam flow rate and pressure. PSEG developed post modification acceptance test procedure S2.PI-SP.ZZ-0001, "Power Ascension Test for HP Turbine and Steam Generator Replacement," to support this work. The purpose of power ascension testing was to validate predicted values for reactor coolant flow rate, pressurizer water level, main steam flow rates and main steam inlet pressure to the high pressure turbine. Power ascension testing was to be initiated at 18% reactor power and was to be performed in conjunction with normal plant startup procedures.

During startup power ascension testing was not implemented as planned and as a result operators did not recognize that all Unit 2 high steam flow protection channels were inoperable until the plant reached 84% power. Operators ultimately identified the condition because the steam flow rate measurement used for protection and indication was outside of acceptable limits, and entered TS 3.0.3 and conducted a plant shutdown.

PSEG completed troubleshooting and determined that a number of engineering deficiencies and incomplete post modification test plan implementation ultimately resulted in the inoperable high steam flow protection channels. This issue is unresolved pending PSEG's completion of its root cause evaluation, inspector review of the root cause evaluation and additional inspector review of design control practices associated with the steam generator replacement project. **(URI 05000311/2008003-04, Salem Unit 2 Steam Flow – Feed Flow Mismatch)**

.2 (Closed) Temporary Instruction 2515/166 – Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)

a. Inspection Scope

The inspectors performed an inspection in accordance with Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage, Revision 1. 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." Specifically, the inspectors verified that the implementation of the modifications and procedure changes were consistent with the actions discussed in PSEG's letters to the NRC, dated December 10, 2007 and February 29, 2008. The February 29, 2008 letter to the NRC included PSEG's response to the draft open items from the NRR audit of corrective actions to address Generic Letter 2004-02, conducted in October 2007.

The inspectors reviewed the technical specifications (TS) and the Updated Final Safety Analysis Report (UFSAR) to verify that required changes to the TS had been approved by the NRC, and that the UFSAR had been or was in the process of being updated to reflect the plant changes. Additionally, the inspectors reviewed a sample of procedures to verify that they were updated to reflect programmatic changes to the facility. The inspectors also reviewed a sample of work orders to verify that specific work had been performed to meet PSEG's GL commitments. Finally, the inspectors discussed details of the containment sump modification with engineers to verify design control of the modification process. Documents reviewed are listed in the attachment. Portions of the TI were performed during the 2006 refueling outage at Unit 2, and the 2007 refueling outage at Unit 1, to verify the physical modifications to the containment sump. The results of those inspections were documented in Inspection Report Nos. 05000311/2006005 and 05000272/2007003.

b. Evaluation of Inspection Requirements

The TI requires 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 PSEG has implemented, or was in the process of implementing, the plant modifications and procedure changes committed to in their GL 2004-02 responses. The inspections previously performed in 2006 and 2007 verified the implementation of the sump screen modifications as related to the GL. During this inspection, the inspectors verified that procedures were updated as related to programmatic controls of potential debris sources, and inspections for containment coating degradation. Additionally, the inspectors verified that evaluations to address downstream effects had been performed on both units. At the time of inspection, inspectors noted that PSEG was still performing chemical effects testing on both units.

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), and identified in PSEG's GL 2004-02 responses, were reviewed and documented in accordance with 10 CFR 50.59. The inspectors also verified that PSEG had obtained NRC approval prior to implementing those changes that required such approval. Specifically, PSEG had obtained NRC approval prior to implementing changes to the methodology for NPSH calculations for the Emergency Core Cooling System (ECCS) pumps. Finally, the inspectors verified that PSEG had either updated, or was in the process of updating, the UFSAR to reflect the required changes in response to GL 2004-02.

Based on the inspectors' review of the hardware modifications, procedure changes, and licensing bases changes, the inspection requirements of the Temporary Instruction are complete and the TI is closed at Salem Units 1 and 2. In a letter to the NRC dated June 26, 2008, as supplemented by letter dated July 10, 2008, PSEG requested an extension for completion of certain corrective actions required under Generic Letter 2004-02. As of this inspection, the remaining activities include completion of strainer performance and chemical bench top testing, and incorporation of the test results into the ECCS NPSH calculations for both units. In a letter dated July 31, 2008, NRR approved PSEG's request to extend the completion date for the remaining testing and analyses required for GL 2004-02 compliance until December 31, 2008. In addition, PSEG has to provide an updated GL 2004-02 supplemental response within 90 days of completing the remaining activities, but not later than March 31, 2009.

The TI-2515/166 inspection results, as well as any results of sampling audits of licensee actions will be reviewed by the NRC staff (Office of Nuclear Reactor Regulation-NRR) as input, along with the Generic Letter 2004-02 responses to support closure of GL 2004-02 and Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance." The NRC will notify PSEG by letter of the results of the overall assessment as to whether GSI-191 and GL 2004-02 have been satisfactorily addressed at Salem Generating Station. Completion of TI-2515/166 does not necessarily indicate that PSEG has finished all testing and analyses needed to demonstrate the adequacy of their modifications and procedure changes. As noted above, PSEG has obtained approval of a plant-specific extension that allows for completion of testing and analyses, as well as any necessary updates to calculations. PSEG will confirm completion of all corrective actions to the NRC. As part of the process described above to ensure satisfactory resolution of GL 2004-02 and GSI-191, the NRC will track all such yet-to-be-performed items identified in the TI-2515/166 inspection reports, track the items to completion, and may choose to inspect implementation of some or all of them.

c. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On July 9, 2008, the resident inspectors presented the inspection results to Mr. Braun. PSEG acknowledged that none of the information reviewed by the inspectors was proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Best, System Engineer – Electrical
R. Chan, Salem Maintenance Superintendant – I&C
V. Ciarlante, Manager Electrical/I&C Design
J. FitzPatrick, Nuclear Technical Supervisor
R. Gary, Radiation Protection Manager
G. Gaydos, Manager Steam Generator Project
J. Giunta, System Engineer – Electrical
L. Gonzalez, Design Engineer – Electrical/I&C
G. Greer, System Engineer – Electrical
R. Halter, Manager Outages
D. Hayman, Salem Maintenance Superintendant – I&C
R. Henriksen, Plant Engineering Manager – Balance of Plant
S. Milligan, I&C Maintenance Supervisor
T. Neufang, Nuclear Technical Supervisor
T. Nolte, Design Engineer – Mechanical/Structural
L. Rajkowski, Manager Design Engineering – Salem
B. Settle, Manager Engineering Response
J. Starceivich, Work Management Cycle Planner
S. Zieglar, Work Week Specialist
E. Villar, Licensing Engineer
K. Mathur, Mechanical Design Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000311/2008003-04	URI	Salem Unit 2 Steam Flow Feed Flow Mismatch (Section 4OA5)

Opened/Closed

05000311/2008003-01	NCV	Salem Unit 2 Loss of All Three Chillers (Section 1R13)
05000311/2008003-02	NCV	Salem Unit 2 Loss of Reactor Vessel Level Indication System (Section 1R15)
05000311/2008003-03	NCV	Salem Unit 2 22 CFCU Valves Mispositioned (Section 1R22)
05000272,311/2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (Section 40A5)

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

OP-SH-108-107-1001, Electric System Emergency Operations and Electric Systems Operator Interface, Revision 1
S1.OP-AB.GRID-0001, Abnormal Grid, Revision 18
SH.OP-AP.ZZ-0027, On-Line Risk Assessment, Revision 14
WC-AA-101, On-Line Work Management Process, Revision 16
WC-AA-107, Seasonal Readiness, Revision 7
NC.CC-DG.ZZ-0003, PRA Weekly Risk Assessment (a)(4) Desktop Guide, Revision 4
OP-SH-101-112-1002, On-Line Risk Assessment, Revision 1
SC.OP-AB.ZZ-0001, Adverse Environmental Conditions, Revision 11
SC.OP-PT.ZZ-0002, Station Preparations for Seasonal Conditions, Revision 11

Drawings

211806	220948	627842	238079	605432	605455

Notifications

20363571	20369892	20366496	20373899	20366190	20365850
20373092	20372684	20372156-9	20369892	20372088	20371363
20372686	20364726				

Orders

60068932	60072761	80094805

Other Documents

2008 Salem Summer Readiness Affirmation letter to Site Vice President dated 5/21/08
Salem Switchyard Readiness 2008 Summer Period from T. Fries dated 4/21/08
WC-AA-107, Seasonal Readiness, Revision 7 completed portions for summer season 2008
Adverse Condition Monitoring and Contingency Plan 08-009, Salem Switchyard Hot Spots, Revision 1
Plant System Readiness Review for Service Water, Service Water Intake Structure (SWIS) and SWIS Ventilation, dated January 18, 2008
Plant System Readiness Review for Salem Auxiliary Building Ventilation, dated January 22, 2008
Plant System Readiness Review for Salem Control Building Ventilation, dated January 22, 2008
Plant System Readiness Review for Salem Switchyard and Transformers, dated December 26, 2007
SPM-2008-007, 2008 Salem Summer Seasonal Readiness Affirmation, dated May 30, 2008
ACM 08-009, Salem Switchyard Hot Spots, Revision 1

Section 1R04: Equipment Alignment

Procedures

S1.OP-SO.CAV-0001, Control Area Ventilation Operation, Revision 33
S1.OP-SO.RM-0001, Radiation Monitoring Systems Operation, Revision 34

S2.OP-IO.ZZ-0011, Control Operability with Unit 2 Defueled, Revision 3
S2.OP-SO.125-0007, 2C 125VDC Bus Operation, Revision 17

Drawings
205248 205348 205332 205342

Notifications
20364182

Other Documents
S-C-CAV-MEE-1285, Control Room Ventilation – Radiological Contaminated Air Intrusion, Revision 0
S-C-ZZ.MDC-1945, Post-LOCA EAB, LPZ, & CR Doses – Alternative Source Term (AST), Revision 0
DE-CB.CAV-0013, Configuration Baseline Documentation Control Area Ventilation System, Revision 4
Tagging Work List 4213853, SG Replacement (Primary Side) 2R16, April 15, 2008

Section 1R05: Fire Protection

Drawings
602537

Notifications
20366022 20365999 20370791 20373417

Orders
60077034

Other Documents
NC.DE-PS.ZZ-0001(Q)-A2-FHA, Salem Fire Protection Report Fire Hazards Analysis, Revision 6
FRS-II-911, Service Water Intake Structure Elevations: 92’ & 112’ Pre-Fire Plan, Revision 2
FRS-II-914, Salem – Unit 1, Unit 2 – Pre-Fire Plan, Outer Penetration Area
FRS-II-453, Salem – Unit 1, Unit 2 – Pre-Fire Plan, Auxiliary Building Ventilation Units

Section 1R07: Heat Sink Performance

Orders
30159599 30157448 30155486

Section 1R08: Inservice Inspection Activities

Procedures
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-836-12, Revision 12, 8/29/08; 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
ER - AP - 335 - 1012, Revision 3, Bare Metal Visual Examination of PWR Vessel Penetrations and Nozzle Safe Ends
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 12; Corrective Action Program (CAP) Procedure
LS - AA - 120, Revision 8; Issue Identification And Screening Process
SH.RA-IS.ZZ-0005(Q)-Revision 6; VT-2 Visual Examination of Nuclear Class 1, 2 and 3 Systems
SH.RA-IS.ZZ-0150(Q) – Revision 8, 10/19/04; Nuclear Class 1, 2, 3 and MC Component Support Visual Examination

<u>Drawings</u>
PSEG dwg. 247393A1719-12, Revision 10: Salem Nuclear Generating Station No. 2 Unit – Reactor Containment Steam Generator Supports
PSEG dwg. 247393A1719-14, Revision 10: Salem Nuclear Generating Station No. 2 Unit – Reactor Containment Steam Generator Supports

<u>Notifications</u>

20366125	20334399	20299708	20331083
20367910	20354066	20263549	20322680
20250549	20354300	20303090	20318470
20265818	20359214	20305952	20318222
20298282	20360336	20350169	20315166
20300320	20299708	20350341	20309819
20300577	20300768	20354066	20300577
20303090	20350169	20354300	20300320
20305952	20350341	20356786	20298282
20309819	20354066	20356795	20365793
20315166	20356786	20331037	20365892
20318222	20356795	20360336	20365150
20318407	20360336	20359214	
20322680	20337886	20354066	
20331083	20337903	20334399	

<u>Orders</u>
DCR 80091019; Replacement of RCS Hot & Cold Leg Piping RTD's and Thermowells
60066694 60060182 60073579 60057022

<u>SGT NCRs Accepted "as-is":</u>
NCR 203
NCR 204

Section XI Repair Replacement Samples:
Notification 20306803, 12/7/06; Weld leak upstream of 2CS61
Notification 20265705, 12/22/05; ASME Leak On Tube At 22SW237
Notification 20346028, 11/24/07
Notification 20346698, 11/24/07; Weld Leak (ASME) Downstream of 2SW341
Notification 20250527, 8/22/05; ASME Leak Downstream of 24SW5

System & Program Health Reports:
System Health Report, Salem Inservice Inspection Program (ISI/CISI/SPT/Snubbers/Repair Programs), 4th Quarter 2007, White

Self Assessments:
FASA Exelon ISI/NDE Procedure Transition Effectiveness, 9/28/08
Boric Acid Corrosion Management and Alloy 600 Program Implementation, 11/23/05; Order 8008641

NDE Examination Reports & Data Sheets
007115, RV Lower Head Flux Thimble Tubing BMV (VT-2)
007116, RV Lower Head Penetration Welds BMV (VT-2)
Includes Photographs: 0182, 0184, 0185, 0186, 0187, 0188, 0189, 0190, 0191, 0192, 0193, 0194, 0195, 0196, 0197, 0198, 0199, 0201, 0202, 0203, 0204, 0205, 0206, 0207, 0208, 0209, 0210, 0211, 0212, 0213, 0214, 0215, 0216, 0217, 0218, 0219, 0220, 0221, 0222, 0223, 0224, 0225, 0226, 0227, 0228, 0229, 0230, 0231, 0232, 0233, 0234, 0235, 0236, 0237, 0238, 0239, 0240, 0241, 7579, 7601, 7605, 7619, 7622, 7623, 7624, 7625, 7626, 7627, 7629, 7630, 7631, 7632, 7633, 7634, 7635, 7636, 7637, 7638, 7639, 7643, 7641, 7644, 7645, 7646, 7647, 7649
010400, 2-PZR-Long-D Longitudinal Weld (UT), 3 pages
011100, 4-PSN-1231-IRS (UT)
031100, 3-CV-1243-7 (UT)
060200, 14-PS-1231-2 (UT)
080010, #24 hot leg thermowell welds BMV (VT-2)
152000, 14-RH-1211-17PS (PT)
167800, 8-SJ-1252-3 (UT)
168200, 8-SJ-1252-9 (UT)
350040, 8-CS-2225-4 (UT)
824800, Penetration-S2-M-29 (VT-G)
074650, RSG #22 Cold Leg Safe End to Elbow (RT)
074650, RSG #22 Cold Leg Safe End to Elbow (UT-pre-service)
082150, RSG #22 Hot Leg Safe End to Elbow (RT)
082150, RSG #22 Hot Leg Safe End to Elbow (UT-pre-service)
381401, RSG #24 Main Steam Nozzle Weld (RT)
330612, RSG #23 Feed Water Nozzle Weld (RT)
330912, RSG #22 Feed Water Nozzle Weld (UT-pre-service)
S2CS-2CS61, 12/8/06, pre-repair
S2CS-2CS61, 12/8/06, pre-repair
S2CS-2CS61, 12/8/06, post-repair – clean
Weld History Record 71352 (Order 60060182), weld ID: S2-IT-472-HB-1, 12/23/05
Multiple Weld History Record 72905 , weld ID: S2-60075716-FW, 1 thru 8, 11/30/07
510821, 11 SG Lower Vertical Support
734260, No. 2 Boron Injection Tank Supports
734270, Regenerative Heat Exchanger Support

734150, No. 21 SG Lower Vertical Support
510825, 12 SG Lower Vertical Support
510830, 13 SG Lower Vertical Support
510835, 14 SG Lower Vertical Support
950000, All Class I Systems (VT-2), NOP/NOT walkdown, 5/12/08

Personnel Qualifications
Certification of Qualification and Qualification Summary, Robert Sistek
Welder Qualification Record, J. Eskesen, 4/2/08

Repair-Replacement Work Orders
DCR 80091019; Replacement of RCS Hot & Cold Leg Piping RTD's and Thermowells
60066694
60060182
60073579
60057022

Calculations:
VTD 328417(001) PSE-06-41 Steam Generator Snubber Elimination, Westinghouse WCAP-16544-P

50.59 Screens & Reviews
DCP 80091019

Calculations
SC-RCP001-01, Overtemperature DT Scaling
VTD Weed Instrument Company, Conax Corporation
70049921
70077209
70052516
70064452

Other Documents
NRC Generic Letter 91-17, 10/17/91; Generic Safety Issue 29, Bolting Degradation or Failure In Nuclear Power Plants
PSEG Ltr. LRN-03-0367, 9/11/2003; 30-Day Response to NRC Bulletin 2003-02, Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundry Integrity, Salem Generating Station Units 1 and 2, Docket Nos. 50-272 AND 50-311, Facility Operating Licenses No.s DPR-70 and DPR-75
PSEG Ltr. LR-N03-0524, 12/23/2003; Response to NRC Bulletin 2003-02, Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundry Integrity, Salem Generating Station Unit 2, Docket No. 50-311, Facility Operating License NO. DPR-75
PSEG Ltr. LR-N04-0297, 7/6/2004; NRC Bulletin 2003-02 Inspection Results, Leakage From Reactor pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundry Integrity, Salem Generating Station Unit 1, Docket No. 50-272, Facility Operating License No. DPR-70
Salem Unit 2 Report S2RFO15, Owner's Activity Report, 1/16/07

Section 1R11: Licensed Operator Requalification Program

Procedures
S2.OP-AB.CW-0001, Circulating Water System Malfunction, Revision 28
S2.OP-AB.LOAD-0001, Rapid Load Reduction, Revision 16
S2.OP-AB.TL-0001, Loss of Main Turbine Lube Oil, Revision 4
S2.OP-AB.TRB-0001, Turbine Trip Below P-9, Revision 14
S2.OP-AB.STM-0001, Excessive Steam Flow, Revision 9
2-EOP-TRIP-0001, Reactor Trip or Safety Injection, Revision 27
2-EOP-TRIP-0003, Safety Injection Termination, Revision 25

Other Documents
SG-0853, Simulator Training Scenario AB.CW, AB.TL, AB.STM, TRIP-3, Revision 0

Section 1R12: Maintenance Effectiveness

Procedures
S2.OP-LR.CA-0008, Leak Rate Test of 2PR1 PORV Accumulator, Revision 2
S2.OP-LR.CA-0009, Leak Rate Test of 2PR2 PORV Accumulator, Revision 3
S2.OP-AR.ZZ-0005, Overhead Annunciators Window E, Revision 17
S2.OP-AB.CA-0001, Loss of Control Air, Revision 15
SC.ER-DG.ZZ-0002, System Function Level Maintenance Rule Scoping vs. Risk Reference, Revision 2
S1.OP-SO.CAV-0001, Control Area Ventilation Operation, Revision 33
S1.OP-SO.RM-0001, Radiation Monitoring Systems Operation, Revision 34
S2.OP-IO.ZZ-0011, Control Operability With Unit 2 Defueled, Revision 3
S2.OP-SO.125-0007, 2C 125VDC Bus Operation, Revision 17

Drawings

205301	205347	249614	249615	24383	21867
205248	205348				

Notifications

20367032	20368007	20373131	20365034	20364182	20372997
20365479	20365869				

Orders

70084683	70084735	70083806	80095636

Other Documents
S-C-CA-MDC-1169, Evaluation of Pressure Decay of PORV Accumulators and Check Valves, Revision 1
Salem Inservice Testing Basis Document, Revision 8
Table 9-2B of Salem Inservice Testing Program Basis Data Sheets – Valves, Revision 2
DE-CB.CA-0014, Configuration Baseline Documentation for Control Air and Station Air Systems, Revision 5
DE-CB.RC-0042, Configuration Baseline Documentation for Reactor Coolant System, Revision 2
S-C-ZZ-MDC-1945, Post-LOCA EAB, LPZ, & CR Doses – Alternative Source Term (AST), Revision 0
DE-CB.CAV-0013, Configuration Baseline Documentation for Control Area Ventilation System, Revision 4

S-C-CAV-MEE-1285, Control Room Ventilation – Radiological Contaminated Air Intrusion, Revision 0
NOS05CAVENT-06, Salem Operations Training Lesson Plan, Control Area Ventilation System

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
SC.OM-AP.ZZ-0001, Shutdown Safety Management Program – Salem Annex, Revision 2
OP-AA-101-112-1002, On-Line Risk Assessment, Revision 2
S2.OP-AB.SF-0001, Loss of Spent Fuel Pool Cooling, Revision 7
S2.OP-SO.SW-0003, 22 Nuclear Service Water Header Outage, Revision 22
SC.MD-FR.CAN-0001, Revision 12
SH.OP-AP.ZZ-0015(Q), Safety Tagging Operations, Revision 20
NC.NA-AP.ZZ-0015(Q), Safety Tagging Program, Revision 19

Notifications
20364796 20364771 20372581 20372594 20371577 20353311
20375596

Orders
70083693

Other Documents
2R16 Refuel Outage ORAM Risk Profile dated February 5, 2008
Contingency Plan for Inventory Control and Shutdown Cooling (2R16 Refueling Outage)
NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, December 1991
Tagging Work List 4222597, 22 SW Hdr Outage (disabled valves) 2R16, April 10, 2008
SGS Unit 1 PRA Risk Evaluation Form for work week 814 (March 30 to April 5, 2008), Revision 2
SGS Unit 1 PRA Risk Evaluation Form for work week 821 (May 18 to 24, 2008), Revision 0
SGS Unit 2 PRA Risk Evaluation Form for work week 821 (May 18 to 24, 2008), Revision 0
TQ-AA-119-0202, Clearance and Tagging Training Program, Revision 3

Section 1R15: Operability Evaluations

Procedures
S1.OP-SO.CAV-0001, Control Area Ventilation Operation, Revision 33
S1.OP-SO.RM-0001, Radiation Monitoring Systems Operation, Revision 34
S2.OP-IO.ZZ-0011, Control Operability with Unit 2 Defueled, Revision 3
S2.OP-SO.125-0007, 2C 125VDC Bus Operation, Revision 17
S1.OP-PT.SW-0007, Service Water Fouling Monitoring Containment Fan Coil Units, Revision 10
MA-AA-1000, Conduct of Maintenance Manual, Revision 7
MA-AA-716-009, Use of Maintenance Procedures, Revision 0
MA-AA-716-011, Work Execution & Close Out, Revision 9
S2.OP-SO.RVL-0001, Reactor Vessel Level Instrumentation System, Revision 18
SC.IC-PT.RVL-0001, RVLIS Level Output Scaling Adjustments and Heat-up Data Collection, Revision 11
VS2.IC-CC.RVL-0001, Train A Reactor Vessel Level Instrumentation System RVLIS-86, Start-up/Check-out Vendor Procedure, Revision 4

2-EOP-LOCA-2, Post LOCA Cooldown and Depressurization, Revision 25
2-EOP-TRIP-4, Natural Circulation Cooldown, Revision 23
2-EOP-CFST-1, Critical Safety Function Status Trees, Revision 25
2-EOP-FRCC-1, Response to Inadequate Core Cooling, Revision 22
2-EOP-FRCI-3, Response to Void in Reactor Vessel, Revision 25
SC.ER-DG.ZZ-0002, System Function Level Maintenance Rule Scoping vs. Risk Reference, Revision 2

Drawings

205248	205348	205242	172571	172572

Notifications

20364182	20365832	20316884	20366122	20363521	20369187
20371034	20370633	20370632	20371850	20372382	20364724
20363924	20364534	20363320			

Orders

30158263	70067760	80091018	70085725	50099455	30668055
30096143	30124096	30145027	30131784	60076704	80086507
70083532	70082359	70082360			

Other Documents

S-C-CAV-MEE-1285, Control Room Ventilation – Radiological Contaminated Air Intrusion, Revision 0
S-C-ZZ.MDC-1945, Post-LOCA EAB, LPZ, & CR Doses – Alternative Source Term (AST), Revision 0
DE-CB.CAV-0013, Configuration Baseline Documentation Control Area Ventilation System, Revision 4
S-C-SW-MDC-1350, Service Water System MODE OPS Analysis, Revision 7
S-C-SW-MDC-1500, Biofouling Monitoring and Trending Calculation, Revision 1
VTD 317348, Salem Units 1 and 2 Reactor Containment Fan Cooler Motor Cooler Reduced Water Flow Evaluation, Revision 2
MPR Associates letter 0108-0807-0359-1, Salem Unit 2 WEKO Seal Backing Ring Evaluation
VTD 314243, RVLIS/Rx Vessel Lvl System, Revision 2
LR-N08-0133, Special Report 311/2008-004, Two Channels of Reactor Vessel Level Instrumentation System (RVLIS) Inoperable Greater Than 7 Days
Fact Finder records documenting results of fact finding interviews of personnel involved with RVLIS scaling
Electronic mail between Westinghouse and PSEG concerning RVLIS indication and operability
Background Information for Westinghouse Owners Group emergency Response Guidelines Generic Issue Reactor Vessel Liquid Inventory System, HP/LP-Rev. 2, dated April 30, 2005

Section 1R18: Plant Modifications

Notifications

20359859	20376021

Other Documents

CC-SH-112-1001, Temporary Configuration Change Implementation T7RM, Revision 0
CC-AA-112, Temporary Configuration Changes, Revision 11
CC-SH-103-1001, CFCU Simplification - Fixed Resistance Control Scheme, Revision 1

Section 1R19: Post-Maintenance Testing

Procedures
MA-AA-716-012, Post Maintenance Testing, Revision 10
NC.MD-AP.ZZ-0050, Maintenance Testing Program Matrix, Revision 10
SH.MD-GP.ZZ-0240, System Pressure Test at Normal Operating Pressure and Temperature, Revision 8
S2.OP-ST.DG-0001, 2A Diesel Generator Surveillance Test, Revision 44
SC.MD-EU.DG-0009, Astro-med Dash 8N Recorder/Equipment Setup for EDG Related Surveillance Testing, Revision 1
SH.MD-EU.ZZ-0001, Crimping Instructions, Revision 8
SH.IC-TI.ZZ-0001, Electronic Soldering/Desoldering, Revision 3
SC.MD-PM.CC-0002, Component Cooling Heat Exchangers # 11, 21, and 22 Internal Inspection, Revision 13
CC-AA-501, PSE&G Nuclear LLC Welding/Brazing Manual, Revision 0
SH.MD-GP.ZZ-0240, System Pressure Test at Normal Operating Pressure and Temperature, Revision 8
MA-AA-716-012, Post Maintenance Testing, Revision 10
NC.MD-AP.ZZ-0050, Maintenance Testing Program Matrix, Revision 10
SC.IC-CC.NIS-0017, N43 Power Range, Revision 4
SC.IC-TI.NIS-0001, Triaxial Cable Crimp-on Connector Installation, Revision 3
SC.IC-TI.NIS-0010, Excore Nuclear Instrument System Detector Assembly, Removal and Installation, Revision 3
SC.IC-PT.NIS-0009, Power Range Detector Post-Installation Electrical Tests, Revision 3
S2.OP-ST.SJ-0005, Inservice Testing Safety Injection Valves Modes 5-6, Revision 16
SC.MD-ST.28D-0001, Preventive Maintenance and 18 Month Surveillance of 28 Volt Station Battery Chargers, Revision 15

Notifications

20364991	20369693	20373219	20373243	20367469	20366717
20366178	20366218	20367183			

Orders

CM980824163	50113422	50113438	50113434	60054063	30126784
60076709	70085158	60076388	60076389	30126880	50103268

Drawings

211347	211357	601241

Other Documents
MOV Diagnostic Test Instructions/Criteria for 21SJ44, dated April 18, 2008
MOV Diagnostic Test Instructions/Criteria for 22SJ44, dated April 18, 2008

Section 1R20: Refueling and Outage Activities

Procedures
S2.OP-PT.CAN-0001, Containment Walkdown, Revision 18
S2.OP-SO.RC-0006,   Revision 28
S2.OP-ST.FHV-0001, Refueling Operations Fuel Handling Building Ventilation, Revision 7
OP-AA-3, Reactivity Management, Revision 0
OP-AA-300-1520, Reactivity Management – Fuel Handling, Storage and Refueling, Revision 2

SC.RE-AP.ZZ-0049, Salem Conduct of Fuel Handling, Revision 1
SC.RE-FR.ZZ-0019, Refueling, Revision 11
S2.OP-ST.CAN-0007, Refueling Operations – Containment Closure, Revision 24
OP-SH-108-108-1001, Hope Creek & Salem Drywell/Containment Close-out, Revision 0
S2.OP-IO.ZZ-0009, Defueled to Mode 6, Revision 22
SC.MD-FR.CAN-0001, Outage Equipment Hatch Installation, Removal, Seal Replacement and Door Manipulation For Containment Closure, Revision 12
S1.OP-AB.CC-0001, Component Cooling Abnormality, Revision 11
S1.OP-SO.CC-0001, Component Cooling System Operation, Revision 17

Notifications

20366022	20365999	20367844	20365479	20364724	20363521
20366717	20365869	20366010	20366583	20367562	20364964
20367562	20359937	20359056	20361139	20365078	20363384
20364325	20364903	20364297	20364296	20364295	

Orders

70084153	70084446	50112367	20113044	60075701	30125141

Drawing
205331

Other Documents
2R16 Refuel Outage ORAM Risk Profile dated February 5, 2008
2R16 Initial Risk Assessment Report dated February 5, 2008
Contingency Plan for Inventory Control and Shutdown Cooling (2R16 Refueling Outage)
Containment Closure Assignments for April 28, 2008
Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000
PSEG Letters LR-N02-010, LR-N03-0086, LR-N03-0136, AND LR-N04-0359 - Movement of Irradiated Fuel
NRC Letter dated November 2, 2004 - Correction to Issuance of Amendment Nos. 263 and 245
1-EOP-APPX-1, Component Cooling Water Restoration, Revision 22
Document No. 32-504914-02, Salem Unit 1 & 2 RV Head Drop Evaluation
OP-AA-108-108, Unit Restart Review, Revision 7

Section 1R22: Surveillance Testing

Procedures
SC.MD-EU.CRN-0013, Fuel Handling Crane Daily Use Inspection, Revision 4
S2.OP-PT.SW-0007, Service Water Fouling Monitoring Containment Fan Coil Units, Revision 16
S2.OP-ST.ZZ-0004, 92 Day Locked Valve Verification, Revision 3
S2.OP-ST.SJ-0020, Periodic Leakage Test RCS Pressure Isolation Valves, Revision 19
S2.OP-ST.AF-0009, Plant Systems - Auxiliary Feedwater, Revision 24
S2.OP-ST.AF-0009, Plant Systems - Auxiliary Feedwater, Revision 25
S2.OP-ST.AF-0006, Inservice Testing Auxiliary Feedwater Valves, Revision 12
S2.OP-ST.AF-0003, Inservice Testing - 23 Auxiliary Feedwater Pump, Revision 43
S2.OP-ST.MS-0003, Steam Line Isolation and Response Time Testing, Revision 16
S2.OP-ST.SJ-0021, RCS Pressure Isolation Leakage Test 2RH1 and 2RH2 Valves, Revision 9

Notifications

20374341	20373860	20372695	20373709	20372862	20372863
20368319	20372729	20374025	20368544	20368862	20369101
20369039					

Order
30145776

Other Documents
OP-AA-108-101-1002, Component Configuration Control, Revision 1
OP-SH-108-103-1001, Locked Equipment Program, Revision 0
OP-AA-108-103, Locked Equipment Program, Revision 2
OP-AA-116-101, Equipment Labeling, Revision 2
HU-AA-104-101, Procedure use and Adherence, Revision 3

Sections 2OS1: Access Control to Radiologically Significant Areas; 2OS2: ALARA Planning and Controls; and 2OS3: Radiation Monitoring Instrumentation

Condition Reports

20364256	20363693	20363678	20363703	20363702	20363693
20363678					

Section 4OA1: Performance Indicator Verification

Other Documents
Salem 1 1Q/2008 Performance Indicators
Salem 2 1Q/2008 Performance Indicators

Section 4OA2: Identification and Resolution of Problems

Notifications
20370007 20371360 20370744

Section 4OA5: Other Activities

Procedures
OP-AA-108-110, Revision 0; EVALUATION OF SPECIAL TESTS OR EVOLUTIONS
SGT Quality Execution Procedure 07.08, Design Change Packages, dated 11/14/2006
SGT Quality Execution Procedure 07.09, Engineering Change Requests, dated 2/28/2006
S2.PI-SP.SG-0001(Q), Revision 1, 5/8/08; 10% Load Rejection Performance Test
S2.PI-SP.SG-0003(Q), Revision 1, 5/8/08; ADFCS Control System Tuning
S2.PI-SP.ZZ-0001(Q) – Revision 11, 5/20/08; Power Ascension Test for HP Turbine and Steam Generator Replacement
SAP 80091187-0020, 6/11/07; Salem Unit 2 Replacement Steam Generator Moisture Carryover Test, Revision 0
NC.NA-AP.ZZ 0059(Q)-Revision 13, 6/22/06; Regulatory Change Determination And 10 CFR 50.59 Review Process
CC-SH-103-1001, Revision 2; Implementation of Configuration Changes
CC-AA-103, Revision 11; Exelon Nuclear Configuration Change Control
LS-AA-104, Revision 5; Exelon Nuclear 50.59 Review Process

SC.MD-EU.CAN-0001(Q), Revision 10, 9/21/06; Inner Equipment Hatch Removal, Seal Replacement and Installation
SC.MD-FR.CAN-0001(Q), Revision 12, 10/12/06; Outage Equipment Hatch Installation, Removal, Seal Replacement and Door Manipulation For Containment Closure
SC.MD-FR.CAN-0002(Q), Revision 0, 10/12/06; Outage Equipment Hatch Ventilation Barrier Installation, Removal, and Closure Test
MA-AA-716-004, Revision 7; Conduct of Troubleshooting
LS-AA-120, Revision 8; Issue Identification and Screening Process
LS-AA-125, Revision 12; Corrective Action Program (CAP) Procedure
S2.OP-IO.ZZ-0006(Q), Revision 34, 3/12/08; Hot Standby to Cold Shutdown

Drawings
PSEG Dwg. 605879-0, Sh.1 of 7, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605879-0, Sh.2, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605879-0, Sh.3, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605879-0, Sh.4, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605879-0, Sh.5, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605879-0, Sh.6, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605879-0, Sh.7, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
PSEG Dwg. 605880-0, Sh.1, Revision 0; Artificial Island Hancock's Bridge, New Jersey, Site Wide Haul Paths
Framatome Dwg. NFPMG-DB-0001, Revision B, 3/18/05; Salem Unit 2 61/19T RSG, General Assembly – Drawing For Pressure Vessel Sizing, SG No. 21-22-23-24
Framatome Dwg. NFPMG-DB-0002, Revision F, 10/26/06; Salem Unit 2 61/19T RSG, General Assembly Internal View, SG No. 21
Framatome Dwg. NFPMG-DB-0006, Revision C, 3/18/05; Salem Unit 2 61/19T RSG, General Assembly Internal View – Parts List - SG No. 21-22-23-24
Framatome Dwg. NFPMG-DB-0007, Revision D, 10/26/06; Salem Unit 2 61/19T RSG, General Assembly External View, SG No. 21
Framatome Dwg. NFPMG-DB-0011, Revision C, 4/1/05; Salem Unit 2 61/19T RSG, General Assembly External View – Detail And Parts List - SG No. 21 – 22 – 23 – 24
Framatome Dwg. NFPMG-DB-0012, Revision B, 3/18/05; Salem Unit 2 61/19T RSG, General Assembly – Drawing List – SG No. 21 – 22 – 23 – 24
Framatome Dwg. NFPMG-DB-0013, Revision C, 3/18/05; Salem Unit 2 61/19T RSG, Channel Head – General View And Details , SG No. 22, 23
Framatome Dwg. NFPMG-DB-0014, Revision C, 3/18/05; Salem Unit 2 61/19T RSG, Channel Head – General View And Details, No. 21, 24
Framatome Dwg. NFPMG-DB-0021, Revision E, 6/1/05; Salem Unit 2 61/19T RSG, Secondary Pressure Vessel Details, SG No. 21 – 22 – 23 – 24
PSEG Dwg. 201193 A 8710-11, Revision 8, 3/28/80; Salem Nuclear Generating Station No. 1 & 2 Units – Reactor Containment Equipment Hatch And Personnel Locks Arrangement, Structural
Trentec Dwg. 31800, Revision 3, 2/29/99; Reactor Containment Building Outage Equipment Hatch, Elevation 140' 0", General Arrangement & Installation

Trentec Dwg. 31801, Revision 4, 2/29/99; Reactor Containment Building Outage Equipment Hatch, Elevation 140’ 0”, General Assembly
Trentec Dwg. 31803, Revision 3, 2/29/99; Reactor Containment Building Outage Equipment Hatch, Elevation 140’ 0”, General Assembly Sections & Details
Trentec Dwg. 31804, Revision 3, 2/29/99; Reactor Containment Building Outage Equipment Hatch, Elevation 140’ 0”, Door Weldment Assembly and Bill of Materials
Framatome Dwg. 1252119E, Revision 2; Salem Replacement Steam Generator 11, Modified Lower Support Column Installation, Sheet 1 of 1
Framatome Dwg. 1257512E, Revision 2; Salem Replacement Steam Generator 12, Modified Lower Support Column Installation, Sheet 1 of 1
Framatome Dwg. 1258218E, Revision 4; Salem Replacement Steam Generator Modified Support Column Assembly
Framatome Dwg. 1252121E, Revision 4; Salem Replacement Steam Generator Modified Lower Support Column Installation Details, Sheet 1 of 2
Framatome Dwg. 1258238 B-02/FTI-22159, Revision 2; Salem Steam Generator Threaded Bolt Fabrication Drawing
Framatome Dwg. 1258239 B-02/FTI-22159, Revision 2; Salem Steam Generator Threaded Bolt Fabrication Drawing
Framatome Dwg. 1258218E, Revision 4; Salem Replacement Steam Generator Modified Support Column Assembly, Sheet 2 of 2
Framatome Dwg. 1252121E, Revision 4; Salem Replacement Steam Generator Modified Lower Support Column Installation Details, Sheet 1 of 2

<u>Design Change Packages(DCPs)</u>
80083663, Revision 0; Steam Generator Supports
80083665, Revision 0; Reactor Coolant System Piping Modification
80083666, Revision 0; Large Bore Piping Modification
80083671, Revision 0; OSG/RSG Rigging & Handling
80083671, Revision 0; Supplement 01, Revision 0; Design input Record
80083671, Revision 0; Supplement 02, Revision 0; Design Change Package
80083671, Revision 0; Supplement 03, Revision 0; Design Verification Checklist
80083671, Revision 0; Supplement 04, Revision 0; Preliminary ALARA Analysis
80083671, Revision 0; Supplement 05, Revision 0; Salem Unit 2 OSG Activity and Load Drop Analysis
80083671, Revision 0; Supplement 06, Revision 0; Qualification of Platform Outside Unit 2 Equipment Hatch for SGRP
80083671, Revision 0; Supplement 07, Revision 0; Salem Unit 2 Polar Crane Evaluation for SGRP
80083671, Revision 0; Supplement 08, Revision 0; OHTS Platform
80083671, Revision 0; Supplement 09, Revision 1; Qualification of Unit 2 Containment Crane and Refueling Canal Wall Concrete for SGRP
80083671, Revision 0; Supplement 10, Revision 0; Mammoet Computation Database
80083671, Revision 0; Supplement 11, Revision 0; No Title
80083671, Revision 0; Supplement 12, Revision 0; Salem Unit 2 Reactor Cavity Decking
80083671, Revision 0; Supplement 13, Revision 0; No Title
80083671, Revision 0; Supplement 14, Revision 0; Stability SPMT
80083671, Revision 0; Supplement 15, Revision 0; Spreader, TLD System
80083671, Revision 0; Supplement 16, Revision 0; Tailing Device/Down Ender
80083671, Revision 0; Supplement 17, Revision 0; Saddles & Skid System HTS
80083671, Revision 0; Supplement 18, Revision 0; Down Ending Device Drawings
80083671, Revision 0; Supplement 19, Revision 0; Cavity Decking Drawings

80083671, Revision 0; Supplement 20, Revision 0; Evaluation of Polar Crane
80083671, Revision 0; Supplement 21, Revision 0; Proposed Well Locations
80083671, Revision 0; Supplement 22, Revision 0; RSG Accelerometers
80083671, Revision 0; Supplement 23, Revision 0; Whiting, Crane inspection Report
80083671, Revision 0; Supplement 24, Revision 0; Whiting Report on Lifting Equipment
80083671, Revision 0; Supplement 25, Revision 0; Haul Route Drawing
80083671, Revision 0; Supplement 26, Revision 0; No Title
80083671, Revision 0; Supplement 27, Revision 0; Secondary Manway Lifting Device
80083671, Revision 0; Supplement 28, Revision 0; Placeholder for Updating Evaluation of Polar Crane
80091015, Revision 0; Evaluation of Unit 2 Control Room Recorder Panel 2RP3 Due to HED Modification
80083522, Revision 0, Supplement 12
80091019, Replacement of RCS Hot & Cold Leg RTD Piping and Thermowells

Engineering Change Requests (ECRs)
ECR-016, dated 5/31/2007
ECR-021, dated 6/18/2007
ECR-026, dated 9/07/2007
ECR-029, dated 6/21/2007
ECR-030, dated 10/12/2007
ECR-032, dated 6/29/2007
ECR-037, dated 7/17/2007
ECR-038, dated 9/07/2007
ECR-039, dated 8/9/2007
ECR-053, dated 9/18/2007
ECR-057, dated 9/18/2007
ECR-058, dated 11/02/2007
ECR-061, dated 11/05/2007
ECR-062, dated 11/02/2007
ECR-065, dated 10/19/2007
ECR-072, dated 11/13/2007
ECR-074, dated 11/13/2007
ECR-075, dated 12/03/2007
ECR-078, dated 11/15/2007
ECR-086, dated 1/30/2008
ECR-215, dated 5/4/08
ECR-071, dated 11/1/07
ECR-066, dated 12/14/07
ECR-215, dated 5/30/08

10 CFR 50.59 Safety Evaluations
80083522, Revision 0; Replacement Steam Generators
Westinghouse Electric Company, EVAL-06-27, Revision 2, Salem Unit 2 - Replacement Steam Generator Analyses and Evaluations for NSSS Scope
S-2-RC-MDC-2151, Revision 0, 1/14/08
WCAP 16444-NP, Revision 1, 8/07; Unit 2Replacement Steam Generator Program NSSS Licensing Report

10 CFR 50.59 Screening Evaluations
80083663, Revision 0: Steam Generator Supports

80083665, Revision 0; Reactor Coolant System Piping Modification
80083666, Revision 0; Large Bore Piping Modification
S-2-RC-MDC-2151, Revision 0, 1/14/08

<u>50.59 Applicability Reviews</u>
S-2-RC-MDC-2151, Revision 0, 1/14/08

<u>Alara Plans</u>
No. 49, Rigging and Moving OSG/RSG, Install Shield Plates
No. 28, Insulation Removal, Replacement and Modification
No. 35, Abrasive Pipe End Decontamination
No. 54, ULS/LLS/Temp RCS Supports, Hot Gap Measurements, In Service Leak Tests
No. 30, Scaffolding Installation, Removal, and Modification
No. 55, Inspections/Support/Oversight/RP & Decon Support/QC/Engineering
No. 51, Radiography of the RSG and Associated Piping
No. 37, Secondary Pipe Cutting, Machining, Welding & FOSAR
No. 53, Install/Remove TLD (Temporary Lifting Device)/HTS (Hatch Transfer System)/ RCD (Reactor Cavity Decking) Inside Containment
No. 47, Electrical, Mechanical, & Structural Interferences
No. 33, RSC Pipe Cutting, Machining & Welding
No. 80, OSG Disposal

<u>Audits and Self-Assessments</u>
NOSPA-SA-07-2Q, Steam generator Replacement Project, Design Change Packages, dated 6/8/2007

<u>Calculations</u>
VTD 328295 (001) Salem Unit 2 RSG – OSG Comparison
PSEG Calc. No. 651-1958, Revision 1; Evaluation of Haul Path and Underground Utilities for SGRP, 10/26/96
PSEG Calc. No. 6A5-2172, Revision 2; Site Wide Haul Path Analysis, Salem and Hope Creek, 4/26/07
PSEG Calc. No. S-C-ZZ-MDC-1920, 6/25/02; Fuel Handling Accidents Occurring in Fuel Handling Bldg. & Containment – AST Analysis for Relaxation of Containment Integrity
PSEG Calc. No. S-C-ZZ-MDC-1920, Revision 3, 4/3/03; Fuel Handling Accident
SGT Calculation 38570-CALC-C-013, Revision 1, 11/30/06; Qualification of Unit 2 Containment, Crane, and Refueling Canal Wall Concrete for SGRP
SGT Calculation 38570-CALC-C-011, Revision 1, 12/13/06; Salem Unit 2 Polar Crane for the SGRP
Calc. 6A5-2172, Revision 2, 4/26/07; Site Wide Haul Path Analysis, Salem And Hope Creek
Calc. S-C-ZZMDC-1920, Revision 3, 3/18/03; Fuel Handling Accident

<u>PSEG Engineering Orders</u>
80091187

<u>PSEG Notifications</u>

20362909	20367429	20367458	20367477	20367786
20367358	20367438	20367471	20367524	20367798
20367368	20367441	20367472	20367590	20367838
20367378	20367454	20367473	20367647	20367877
20367427	20367457	20367475	20367785	20367890

20367904	20367278	20362668	20364060	20362842
20367908	20367257	20362562	20364007	20362803
20367947	20367241	20362310	20363987	20362775
20367962	20367220	20362012	20363927	20362756
20367990	20367216	20361770	20363909	20362732
20367999	20367205	20361562	20363885	20362731
20368047	20366967	20361533	20363802	20362704
20368167	20366964	20361484	20363817	20362602
20368244	20366956	20360991	20363702	20362577
20368263	20366944	20360929	20363693	20362523
20368362	20366911	20360991	20363594	20362402
20368431	20366867	20360929	20363590	20362375
20368445	20366854	20360610	20363510	20362366
20368500	20366850	20360267	20363504	
20368511	20366849	20360257	20363393	
20368606	20366845	20360086	20363317	
20368798	20366788	20360257	20363261	
20362887	20366769	20360086	20363216	
20363329	20366737	20359961	20363214	
20361770	20366695	20359340	20363190	
20361096	20366622	20357408	20363189	
20360610	20366584	20357124	20363032	
20363329	20366504	20356712	20363027	
20362887	20366487	20356170	20363026	
20362909	20366479	20353764	20363025	
20357408	20366475	20352849	20363024	
20351345	20366462	20352848	20363021	
20350725	20366396	20351637	20363010	
20355393	20366379	20351298	20363000	
20356989	20366333	20351290	20362998	
20357074	20366310	20350725	20362994	
20355603	20366131	20350063	20362993	
20355444	20366109	20348361	20362991	
20354615	20366108	20348346	20362985	
20354614	20366045	20348311	20335382	
20354386	20366035	20348278	20324867	
20354361	20366020	20348270	20324119	
20354303	20365892	20364777	20302600	
20354276	20365891	20364755	20301217	
20359025	20365793	20364732	20360142	
20358957	20365768	20364689	20362985	
20359056	20365629	20364650	20362975	
20367305	20365671	20364604	20362951	
20367333	20365584	20364547	20362950	
20367350	20365624	20364513	20362946	
20367354	20362912	20364445	20362911	
20367371	20362909	20364363	20362968	
20367291	20362698	20364301	20362841	

20362284	20361090	20358418	20355188	20362758
20362271	20361066	20358475	20355004	20360703
20362263	20361040	20358030	20351121	20360086
20362204	20361016	20357976	20354702	20367371
20362193	20360978	20358003	20354676	20367241
20362188	20360930	20357627	20354614	20365793
20362156	20360921	20357618	20354361	20363329
20362132	20360905	20357124	20354304	20366976
20361860	20360779	20357117	20354303	20353791
20361848	20360958	20357073	20354276	20353786
20361813	20360875	20357045	20354190	20353764
20361806	20360792	20356989	20354102	20351637
20361794	20360786	20356916	20353791	20351345
20361656	20360779	20356782	20353717	20350725
20361649	20360721	20356625	20353205	20347201
20361587	20360667	20356365	20352901	20347837
20361501	20360653	20356361	20352829	20365892
20361491	20360483	20356315	20351345	20365891
20361478	20360433	20356300	20350601	20365892
20361474	20360411	20356216	20350035	20366584
20361469	20360037	20356213	20350016	20366695
20361460	20360267	20356152	20349958	20366911
20361457	20369989	20356112	20349916	20365891
20361447	20359858	20356093	20349864	20352829
20361441	20359832	20356090	20349443	20368606
20361348	20359787	20355549	20349366	20367838
20361316	20359437	20355541	20348345	20360985
20361211	20359317	20355505	20347837	20361096
20361210	20359305	20355446	20347429	20362968
20361197	20359203	20355357	20347426	20365415
20361127	20359176	20355338	20347201	20367402
20361096	20356989	20355322	20346210	20369574
	20358752	20355300	20367402	20361770

SGT Nonconformance Reports (NCR)

222	188	223	125	160
114	189	224	126	161
113	190	225	127	162
112	191	226	128	163
136	193	227	129	164
164	194	228	130	165
165	195	229	131	166
214	196	230	132	167
218	197	231	133	168
220	198	232	134	169
211	199	233	135	170
205	200	234	136	171
206	201	235	137	172
21	202	236	138	173
220	203	237	139	174
202	204	238	140	175
153	205	239	142	250
219	206	240	144	101
234	207	241	145	102
128	208	242	146	103
163	209	243	147	104
109R1	211	244	148	105
176	212	245	149	106
177	213	246	150	107
178	214	115R1	151	109R1
179	215	116	152	110
180	216	117	153	111R1
181	217	118	154	112
183	218	119	155	112R1
184	219	120	156	113R1
185	220	121	157	114R1
186	221	122	158	197
187	222	124	159	

Evaluations

Altran Solutions Technical Report No. 08-0233-TR-001, Revision 0; Failure Analysis of Steam Generator Support Bolts (OSG)

SGT Apparent Cause Evaluation – Failure of Replacement Steam Generator (RSG) Lower Support Pad Holddown Bolts During RSG Installation Activities, SGT NCR-219

PSEG Supplier Surveillances

PSEG Source Verification Report No.: VS-2005-015; Steam Generator Team, LLC, Vendor No. 23367, P.O. No. 4500245261, 11/10/06

PSEG Source Verification Report No.: VS-2007-011; Steam Generator Team, LLC, Vendor No. 23367, P.O. No. 4500245261, 6/11/07

PSEG Source Verification Report No.: VS-2007-010; Steam Generator Team, LLC, Vendor No. 23367, P.O. No. 4500245261, 5/16/07

PSEG Source Verification Report No.: VS-2007-009; Steam Generator Team, LLC, Vendor No. 23367, P.O. No. 4500245261, 4/30/07
PSEG Source Verification Report No.: VS-2007-008; Steam Generator Team, LLC, Vendor No. 6155, P.O. No. 4500245261

Framatome Manufacturing Nonconformances
03/02001, Revision 00
03/02002, Revision 00
03/02005, Revision 00
03/02005, Revision 01
03/02006, Revision 00
03/02007, Revision 01
03/02008, Revision 01
03/02010, Revision 00
03/02011, Revision 00
03/02012, Revision 00
03/02013, Revision 01
04/02001, Revision 00
04/02002, Revision 00
04/02004, Revision 00
04/02004, Revision 01
04/02004, Revision 02
04/02005, Revision 00
04/02006, Revision 00
04/02009, Revision 00
04/02009, Revision 01
04/02010, Revision 00
04/02011, Revision 00
04/02012, Revision 00
04/02013, Revision 00
04/02014, Revision 00

NDT Procedures
SGT QEP 12.06, Revision 3, 12/17/07; Radiographic Examination (ASME)

Steam Generator Tube Inspections
Areva Procedure 54-ISI-408-00, 11/14/06; Pre-Service Eddy Current Inspection of Salem Unit 2 Steam Generators
Areva Report 51-9044781-001, Revision 0, 2/19/08; Technical Summary of Salem Unit 2 Replacement Steam Generator Pre-Service Eddy Current Inspection, January/February 2007
Safety Document Change Notice (SDCN), 30-9039928-001, Chalon Salem Unit 2 RSG PSI, 1/16/07
AREVA Report 51-9044781-000; Technical Summary of Salem Unit 2 Replacement Steam Generator Pre-Service Eddy Current Inspection, January/February 2007

Shop Hydrostatic Test Reports
Hydrotest Report – Primary: Unit SA305, 2/15/06
Hydrotest Report – Primary: Unit SA306, 6/7/06
Hydrotest Report – Primary: Unit SA307, 1/9/07
Hydrotest Report – Primary: Unit SA308, 3/19/07
Hydrotest Report – Secondary : Unit SA305, 6/6/06
Hydrotest Report – Secondary: Unit SA306, 6/20/06
Hydrotest Report – Secondary: Unit SA307, 3/6/07
Hydrotest Report – Secondary: Unit SA308, 3/19/07

Vendor Documents
Framatome Procedure TMHSSA/NGV0002, Revision C, 1/23/06; In Shop Hydrostatic Test of Secondary Part of Salem Unit 2 Replacement Steam Generators
Framatome Procedure TMHSSA/NGV0001, Revision C, 1/23/06; In Shop Hydrostatic Test of Primary Part of Salem Unit 2 Replacement Steam Generators

Mammoet Document No. 6000044405-M03-00-1, 6/20/07; Root Cause Analysis Spooling Arm Incident CWS650-2, Containerized Winch System
Mammoet Test Procedure CWS-50, Appendix E, Static Load Test Report, 24 April 2007
Mammoet Test Procedure CWS-50, Appendix I, Static Load Test Report TLD
Mammoet Document No. 601341-E-X03, Containerized Winch System (CWS-50) Test Procedure, 19 March 2007
VTD 900013(002), AREVA Salem Unit 2 RSG – Section 1: Introduction and General, AREVA Document BUCRSA/NGV 1984
VTD 328295, AREVA, Salem Unit 2 OSG-RSG Comparison
VTD 900013(1) Framatome Design Report (DR)328295, AREVA, Salem Unit 2 OSG-RSG Comparison, 51-5047612-02
AREVA Document BUCRSA/NGV 2001
AREVA Document NFPMG DC 0022, Revision C
AREVA Document BUERSA/NGV0002, Revision B
AREVA Document BUCRSA/NGV 1987, Revision G
AREVA Document NFPMG DC 0023, Revision B
VTD 900013(008), AREVA Salem Unit 2 RSG – Section 7: Tube Bundle, AREVA Document BUCRSA/NGV 1990
VTD 900169(000), AREVA Salem Unit 2 61/19T RSG – Water Level Stability Analysis, NFPMG DC 13
VTD 900183, AREVA Salem Unit 2 RSG – Allowable Tube Degradation, Regulatory Guide 1.121 Analysis, 4/29/07
VTD 900013(005), 5/17/07; AREVA Salem Unit 2 RSG – Section 4: Lower Assembly, Including Channel Head, Tubesheet, Divider Plate, Lower Shell; AREVA Document UCRSA/NGV 1987
VTD 900182(000), AREVA Salem Unit 2 RSG – Operating Parameters, AREVA NFPMG DC6
VTD 900157(000), AREVA Salem Unit 2 RSG – Equipment Specification, AREVA NFPMG DC1
VTD 900166(001), AREVA Salem Unit 2 RSG – Feedwater System Thermohydraulic Analysis
VTD 900178(001), AREVA Salem Unit 2 RSG – Primary Side Sizing Calculation Report, AREVA NFPMG DC22
VTD 900179(001), AREVA Salem Unit 2 RSG – Secondary Side Sizing Calculation Report, AREVA NFPMG DC23
VTD 900180(001), AREVA Salem Unit 2 RSG – Vibration Analysis of the Tube Bundle (Linear), AREVA NFPMG DC24
VTD 900184(001), AREVA Salem Unit 2 RSG – AVB Mechanical Integrity Analysis, AREVA NFPMG DC28
VTD 900994(001), Pre-service Eddy Current Inspection of Salem Unit 2 Steam Generators, AREVA NP Number 54-ISI-408-00, 11/14/06
VTD 900399, 51-9032150-000, Salem Unit 2 Replacement SG Pre-Service Inspection Degradation Assessment, 1/11/07
VTD 900186(001), 5/17/07, Salem Unit 2 RSG – 61/19T RSG – Loose Part Trapping Systems Mechanical Integrity, AREVA NFPMG DC30
VTD 324204, Trentec Design Report, Outage Equipment Hatch, Salem Nuclear Generating Station Units 1 & 2, Public Service Electric & Gas Company, Report No. DR98031, Revision 2, 4/5/99
VTD 900845, Main Line Engineering, Salem Equipment Hatch Equivalent Closure Device Specification, 1/30/08
VTD 901431, Westinghouse Calculations RSG LL Support, PSEG Vendor Number PSE-08-33, Vendor Code W120

Work Packages
SGT Work Package No. 1030, Install Hatch Transfer System
SGT Work Package No. 1032, Install Outside Lift System
SGT Work Package No. 1033, Install Foundation and Platform for the Outside Hatch Transfer System
SGT Work Package No. 1035, Install Temporary Lifting Device (TLD)
SGT Work Package No. 1050, Install Temporary Power Inside Unit 2 Containment
SGT Work Package No. 1060, Install Reactor Cavity Deck
SGT Work Package No. 2570A, Revision 0, Remove Original Steam Generator OSG 21
SGT Work Package No. 2580A, Revision 0, Transfer OSG 21 To Temporary Storage Facility
SGT Work Package No. 3030A, Revision 1, Transport Replacement Steam Generator 21 From Storage To Containment
SGT Work Package No. 3040A, Revision 0, Install RSG-21
SGT Work Package No. 5030, Revision 0, Remove Hatch Transfer System
SGT Work Package No. 3065C, RCS Machining and Welding SG-23
SGT Work Package No. 4520A, RSG 21 Hot Gap Measurements
SGT Work Package No. 4520B, RSG 22 Hot Gap Measurements
SGT Work Package No. 4520C, RSG 23 Hot Gap Measurements
SGT Work Package No. 4520D, RSG 24 Hot Gap Measurements
SGT Work Package No. 3055D, SG 24 Install Lower Lateral Supports/Remove Temporary SG Supports
SGT Work Package No. 3055C, SG 23 Install Lower Lateral Supports/Remove Temporary SG Supports
SGT Work Package No. 3055B, SG 22 Install Lower Lateral Supports/Remove Temporary SG Supports

Specifications
Detailed Design Specification No. S-2-RC-MDS-0397, Revision 4, Replacement Steam Generator For Salem Generating Station Unit 2

Welding Procedure Specifications (WPS) and Procedure Qualification Records(PQR)
Weld Data Card: RCS Hot Leg Elbow to RSG Nozzle Safe End (SG 23), 4/26/08
Weld Data Card: RCS Hot Leg Elbow to RSG Nozzle Safe End (SG 21), 4/28/08
Weld Data Card: RCS Hot Leg Elbow to RSG Nozzle Safe End (SG 23), 4/26/08
Weld Data Card: RCS Hot Leg Elbow to RSG Nozzle Safe End (SG 22), 4/26/08
Weld Data Card: RCS Hot Leg Elbow to RSG Nozzle Safe End (SG 24), 4/23/08

NDE Data Sheets
DCP 80091019: LP: WP3066A, RSG 21, RTD Prep Surface (Cold Leg), 4/11/08
DCP 80091019: LP: WP3066A-MK-006, Existing Pipe Prep Hot Leg, 4/7/08
DCP 80091019: LP: WP3066A-MK-007, Existing Pipe Prep Hot Leg, 4/7/08
DCP 80091019: LP: WP3066A-MK-008, Existing Pipe Prep Hot Leg, 4/7/08
DCP 80091019: LP: FW-1 RTD/Cold Leg Root Pass
DCP 80091019: LP: FW-1 RTD/Cold Leg Final PT
DCP 80091019: LP: FW-2 Root Pass
DCP 80091019: LP: FW-3 Root Pass
DCP 80091019: LP: FW-4 Root Pass
DCP 80091019: LP: RTD Final (Hot Leg), FW-2
DCP 80091019: LP: RTD Final (Hot Leg), FW-3
DCP 80091019: LP: RTD Final (Hot Leg), FW-4

Miscellaneous

Salem Generating Station, Steam Generator Replacement Project, Weekly Staff Meeting Notes, 1/8/08

SGT Engineering Turnover Meeting Notes, 1/31/08

ASME, NQA-1-2004, Subpart 2.15 Quality Assurance Requirements for Hoisting, Rigging and Transportation of Items for Nuclear Power Plants

Whiting ltr. Dated 12/11/06 to SGT; Public Service Electric & Gas Salem Nuclear Generating Station, Unit 2, 230/35 Ton Reactor Building Polar Gantry Crane S/N 10005 (Built on Whiting Reqn. 84305-09, 1970)

SGT Engineering Turnover Meeting Notes, 3/5/08 Days to 3/5/08 Nights

PSEG ltr. LR-N08-0054, 3/5/08; SALEM GENERATING STATION – Unit 1 and Unit 2 FACILITY OPERATING LICENSE NOS. DPR 70 and DPR-75, NRC DOCLEK NOS. 50-272 and 50-311

Whiting Calculation 75920-0054, 9/24/96; 230/45 TON REACTOR BUILDING POLAR CRANE (460 Ton Construction Lift Capacity, S/N 10004

Whiting Services, Inc., Crane inspection Report, 10/16/06

Westinghouse letter PSE-08-33, Revision 1, 4/24/08; Transmittal of Information for Salem Unit 2 – Items That Must Be Completed Before Entering Modes 6 through 4

Apparent Cause Evaluation, Old Steam Generator (OSG) Support Bolt Failure (70082355)

PORC Meeting Minutes, Meeting Number S2008-007, 4/25/08

PORC Meeting Minutes, Meeting Number S2008-008, 4/28/08

CMTR – RSG Bolts: TAG-SRM-0164-0686, 4/23/08

PSEG ltr. SGR-08-0196, 6/10/08; Review of Apparent Cause Evaluation for Salem Unit 2 Replacement Steam Generator Lower Lateral Support Bolts

PSEG ltr. SGR-07-0507, 8/27/07; Third Party Review of The Rigging And Handling Work Packages, 8/27/07

PSEG ltr. SGR-07-0564, 9/28/07; Transmittal of SGT Rigging And Handling Work Packages

PSEG ltr. SGR-07-0615, 10/24/07; Third Party Review of Whiting Report Including The Finite Element Analysis

PSEG ltr. SGR-07-0067, 2/8/07; Third Party Review of The Haul Route Analysis

PSEG ltr. SGR-07-0147, 3/20/07; Transmittal of Resolution to 3rd Party Haul Path Review Comments

PSEG ltr. LR-N07-0288, 11/19/07; Correction to Amendment 263/245 Safety Evaluation Report

PORC Meeting Minutes, Meeting Number S2008-001, 2/5/08

PSEG ltr. SGR-08-0189, 5/23/08; Transmittal of NCR 219 and ECR-215 For SG LLS Blocks To Westinghouse for Analysis Verification – Salem Unit 2

Steam Flow/Feed Flow Mismatch

Procedures

HU-AA-1211, Briefings – Pre-job, Heightened Level of Awareness, Infrequent Plant Activity and Post-job Briefings

HU-AA-104-101, Procedure Use and Adherence, Revision 3

OP-AA-108-110, Evaluation of Special Tests or Evolutions, Revision 0

S2.PI-SP.ZZ-0001, Power Ascension Test for HP Turbine and Stm Gen Replacement, Revisions 4, 6, 8 -11

SC.RE-RA.ZZ-0004, Statepoint Data Collection, Revision 19

SC.SE-DG.ZZ-0002, Statepoint Data Processing for I&C Procedures, Revision 1

S2.RE-Ra.ZZ-0011, Tables, Revision 245

S2.OP-DL.ZZ-0003, Control Room Readings – Modes 1-4, Revision 1

S2.OP-DL.ZZ-0003, Control Room Readings – Modes 1-4, Revision 2

S2.OP-AR.ZZ-0006, Overhead Annunciators Window F, Revision 13
S2.OP-AR.ZZ-0007, Overhead Annunciators Window G, Revision 43
SC.DE-TS.ZZ-1904, Instrument Setpoint Calculations, Revision 1

Notifications

20373585	20369267	20369574	20372502	20369724	20352829
20369574	20369686	20369881	20369779	20370764	20371567
20372115					

Orders

80083522	70085368	70085441	70085314	70085358	70085444

Other Documents
SC-CN007-01, Salem Unit 1, 2 Steam Generator S.I. Initiate, Steam Flow Ind & Rec, Revision 1
SC-CN007-02. Salem Unit 2 Steam Flow Computerized Scaling, Revision 5B
SC-CN007-02. Salem Unit 2 Steam Flow Computerized Scaling, Revision 5 Final
SC-MS002-01, Turbine Inlet Pressure Scaling/Uncertainty Calculation, Revision 11
SC-RCP001-04, Overpower ΔT / Overtemperature ΔT Uncertainty Calculation, Revision 1
Prompt Investigation U2 Steam Flow/Feed Flow Mismatch
PSE-08-47, Westinghouse Letter to PSEG re: Transmittal of Information for Salem Unit 2 Hot Zero Power Steamline Break Evaluation with Relaxed High Steam Flow Setpoint, dated May 15, 2008
PSE-08-48, Westinghouse Letter to PSEG re: Transmittal of Information for Salem Unit 2 Increased High Steam Flow Setpoint – Impact on Steamline Break Mass/Energy Release Analyses, dated May 23, 2008, Revision 2
SDE-07-0005, PSEG Internal Memo re: NUCP 80083522 Salem 2 Steam Generators Replacement/Key Parameters Values for Scaling/Uncertainty Calculations, Dated May 1, 2007
Complex Troubleshooting Procedure for Salem Unit 2 Steam Flow/Feed Flow Mismatch
OpEval 08-030, Salem Unit 1 Steam Flow/Feed Flow Mismatch, Revision 0
NOS05ADFWCS-07, Operations Training Lesson Plan for Advanced Digital Feedwater Control System
VTD 320367, PSEG Salem Units 1&2 – ADFCS Stm Flow, Stm Press, FW Header Press, Revision 3
VTD 328295, Salem Unit 2 RSG – OSG-RSG Comparison, Revision 1
DE-CB.RCP-0038, Design Basis Documentation for Reactor Protection System, Revision 2
WCAP-16444-NP, Salem Unit 2 Replacement Steam Generator Program NSSS Licensing Report, Revision 1

Calculations
10080-DSC-6762, Sodium Tetraborate Basket Evaluation, Rev. 0

Procedures
CC-AA-102, Design Input and Configuration Change Impact Screening, Revision 15
CC-AA-102-1001, Design Inputs and Impact Screening – Implementation, Revision 3
CC-AA-6006, Monitoring the Performance of Service Level 1 Coating Systems for Salem Generating Units 1 & 2, Revision 0

Orders

80085414	70041422

<u>Other Documents</u>

Letter from PSEG to U.S. NRC: 90-day Response to Generic Letter 97-04 - Assurance of Sufficient NPSH for ECCS and Containment Heat Removal Pumps, Units 1 and 2, LR-N97-0815

Letter from PSEG to U.S. NRC: License Amendment Request LAR S07-05 Revision to Licensing Basis – NPSH Methodology for ECCS Pumps, dated 08/15/2007

Letter from PSEG to U.S. NRC: Supplemental Response to NRC Generic Letter 2004-02, dated 02/29/2008

Letter from U.S. NRC to PSEG: Approval of Request for Extension of Completion Date for Generic Letter 2004-02 Corrective Actions, dated 12/21/2007

Letter from U.S. NRC to PSEG: Issuance of Amendment RE: Revision to Licensing Basis – Net Positive Suction Head Methodology for Emergency Core Cooling System Pumps, dated 11/15/2007

Letter from PSEG to U.S. NRC: Request for Extension To Complete Generic Letter 2004-02 Testing, dated June 26, 2008

Letter from U.S. NRC to PSEG: Draft Request for Additional Information, dated 06/30/2008

Letter from PSEG to U.S. NRC: Response to NRC Request for Information Regarding PSEG Request for Extension to Complete Generic Letter 2004-02 Testing, dated July 10, 2008

S-C-RHR-MDC-2089, Long-Term Wear Effects Evaluation in Support of Resolution of GSI-191, Revision 3

S-C-RHR-MEE-1883, GSI-191 Downstream Effects – Flow Clearances, Salem Units 1 and 2, Revision 1

SCN 06-009, UFSAR Update for Installation of new Containment Sump Screen Configurations in Unit 1 and Unit 2, Revision 0

SCN 07-028, UFSAR Update for NSPH available for the RHR Pumps in Unit 2, Revision 0

VTD 901479, Post LOCA Fuel Deposition Analysis in Support of GSI-191, Revision 0

LIST OF ACRONYMS

ADFCS	Advanced Digital Feedwater Control System
ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAACS	Control Area Air Conditioning System
CAP	Corrective Action Program
CCW	Component Cooling Water
CFCU	Containment Fan Coil Unit
CFR	Code of Federal Regulations
CMTR	Certified Material Test Report
CR	Condition Report/Notification
CRE	Control Room Envelope
CREACS	Control Room Emergency Air Conditioning
CW	Circulating Water
DCP	Design Change Package
DLA	Dynamic Learning Activity
ECCS	Emergency Core Cooling Systems
EDG	Emergency Diesel Generator
ESOC	Electrical Systems Operations Center
FMS	Fundamentals Management System
RW	Feedwater
GL	Generic Letter
GSI	Generic Safety Issue
HX	Heat Exchanger
ISI	In-Service Inspection
LCA	Loss of Control Air
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MS	Main Steam
NCV	Non-cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
Notification	Condition Report
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OSG	Old Steam Generator
OSP	Offsite AC Electrical Power
PARS	Publicly Available Records
PI	Performance Indicator
PJM	PJM Interconnection, Inc.
PORC	Plant Operations Review Committee
PORV	Pilot Operated Relief Valve
PSEG	Public Service Enterprise Group Nuclear LLC
PSV	Pressurizer Safety Valve
PT	Liquid Penetrant Testing
PZR	Pressurizer
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RSG	Replacement Steam Generator

RT	Radiographic Testing
RVLIS	Reactor Vessel Level Indication System
S2R16	Sixteenth Refueling of Salem Unit 2
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SG	Steam Generator
SGRP	Steam Generator Replacement Project
SJ	Safety Injection
SPAV	Switchgear and Penetration Area Ventilation
SPT	Station Power Transformer
SSC	Structure, System, and Component
SW	Service Water
SWIS	Service Water Intake Structure
TCCP	Temporary Configuration Change Package
TI	Temporary Instruction
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
U.S. NRC	United Sates Nuclear Regulatory Commission
UT	Ultrasonic Testing
VT	Visual Testing
WCD	Work Control Document
WO	Work Order
WP	Work Package
WPS	Weld Procedure Specification