November 8, 2006

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

SUBJECT: SALEM NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000272/2006004 and 05000311/2006004

Dear Mr. Levis:

On September 30, 2006, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Salem 1 & 2 reactor facilities. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 5, 2006, with Mr. C. Fricker and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The report documents two NRC-identified findings of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any non-cited violation in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Salem Nuclear Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mel Gray, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000272/2006004 and 05000311/2006004
w/Attachment: Supplemental Information

cc w/encl:
T. Joyce, Site Vice President - Salem
D. Winchester, Vice President Nuclear Assessments
W. F. Sperry, Director Business Support
C. J. Fricker, Salem Plant Manager
J. J. Keenan, Esquire
M. Wetterhahn, Esquire
L. Peterson, Chief of Police and Emergency Management Coordinator
P. Baldauf, Assistant Director, Radiation Protection and Release Prevention, State of New Jersey
K. Tosch, Chief, Bureau of Nuclear Engineering, NJ Department of Environmental Protection
H. Otto, Ph.D., DNREC Division of Water Resources, State of Delaware
Consumer Advocate, Office of Consumer Advocate
N. Cohen, Coordinator - Unplug Salem Campaign
W. Costanzo, Technical Advisor - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

<u>Distribution w/encl</u>:

S. Collins, RA
M. Dapas, DRA
M. Gray, DRP
B. Welling, DRP
D. Schroeder, DRP, Senior Resident Inspector
H. Balian, DRP, Resident Inspector
K. Venuto, DRP, Resident OA
B. Sosa, RI OEDO
H. Chernoff, NRR
D. Collins, NRR
S. Bailey, NRR
T. Valentine, NRR
Region I Docket Room (with concurrences)
ROPreports@nrc.gov

DOCUMENT NAME: C:\FileNet\ML063130529.wpd

SISP Review Complete: <u>MG</u> (Reviewer's Initials)

After declaring this document "An Official Agency Record" it **will/~~will not~~** 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					
NAME	DSchroeder/MG for		MGray/MG					
DATE	11/07/06		11/07/06					

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	50-272, 50-311
License Nos:	DPR-70, DPR-75
Report No:	05000272/2006004 and 05000311/2006004
Licensee:	Public Service Enterprise Group Nuclear LLC
Facility:	Salem Generating Station, Units 1 & 2
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	July 1, 2006 through September 30, 2006
Inspectors:	J. Daniel Orr, Senior Resident Inspector D. L. Schroeder, Senior Resident Inspector H. Balian, Resident Inspector J. G. Schoppy, Jr., Senior Reactor Inspector S. M. Pindale, Senior Reactor Inspector T. L. O'Hara, Reactor Inspector M. P. Snell, Reactor Inspector S. C. McCarver, Project Engineer R. Nimitz, Senior Health Physicist
Approved By:	Mel Gray, Chief Projects Branch 3 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000272/2006004, 05000311/2006004; 07/01/2006 - 09/30/2006; Salem Nuclear Generating Station Units 1 and 2; Maintenance Effectiveness, Event Followup.

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to accomplish maintenance in accordance with procedures. PSEG maintenance personnel omitted procedure steps to adequately tighten or properly lock a locknut on the 22 service water strainer during preventive maintenance. Consequently, the 22 service water strainer motor tripped due to increased strainer basket internal interference after it was returned to service.

  The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone, and it affected the cornerstone objective. Unavailability of the 22 SWS and SWP increased the likelihood of a loss of service water. This finding also impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, incorrectly performed maintenance degraded both availability and reliability of the 22 SWS and SWP. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 SDP screening and determined that a more detailed Phase 2 evaluation was required to assess the safety significance because the performance deficiency affected two cornerstones. However, the Risk-Informed Inspection Notebook for Salem Nuclear Generating Station does not evaluate loss of service water initiating events. Therefore, an NRC Region 1 Senior Reactor Analyst (SRA) conducted a Phase 3 analysis and determined the finding was of very low safety significance (Green). The performance deficiency has a cross-cutting aspect in the area of human performance related to the work practices component, because PSEG did not effectively communicate expectations regarding procedure compliance and personnel did not follow procedures. (Section 1R12)

- Green. The inspectors identified a non-cited violation for PSEG's failure to follow Salem Technical Specification 3.4.11.1.b., Structural Integrity. PSEG discovered a leak on the instrument tubing for reactor coolant system loop flow transmitter 2FT416 and did not properly classify and evaluate the leak for operability or structural integrity, or alternatively isolate the affected tubing.

  The finding is more than minor because it affects the Initiating Events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and at power. The inspectors determined that the finding was of very low safety significance (Green) using a Phase 1 screening in Appendix A of Inspection Manual Chapter 0609, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." It is expected that a tubing crack would result in an increase in reactor coolant system (RCS) leakage, and operators would take action prior to exceeding Technical Specification limits for RCS leakage. Therefore, assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function. The performance deficiency has a cross-cutting aspect in the area of problem identification and resolution, related to the corrective action program component, because PSEG did not thoroughly evaluate the condition. (Section 4OA3)

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period at 100 percent (%) power. On July 22, 2006, operators reduced power to 54%, when a lightening storm caused the Hope Creek Generating Station 5015 Red Lion 500 kV transmission line to de-energize. Operators returned Unit 1 to 100% power on the same day. On September 23, 2006, operators reduced power to 85% to conduct main turbine valve testing. Operators further reduced power to 74% when one main turbine stop valve did not reopen after testing. Following repairs to the main turbine stop valve, operators returned Unit 1 to 100% power on September 24, 2006. Unit 1 remained at full power for the remainder of the quarter.

Unit 2 began the period at 100% power. On September 19, 2006, operators began reducing power as part of a planned production coast-down in preparation for a refueling outage. On September 26, 2006, operators manually tripped Unit 2 when the 21 reactor coolant pump seal return flow rate exceeded an abnormal operating procedure limit. Reactor coolant pump seal return flow rates returned to normal after the plant trip, and there was no indication of damage to the reactor coolant pump seals. Operators returned Unit 2 to power operation on September 27, 2006. Unit 2 ended the quarter at 85% power consistent with the planned production coast-down.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope (1 sample)

The inspectors reviewed PSEG's response to adverse weather conditions during late June and early July 2006. Specifically, the inspectors reviewed PSEG's response to increased Delaware River detritus loading following heavy rainfall the week of June 26, 2006. The inspectors reviewed Units 1 and 2 service water system performance and impact on supported systems. The inspectors observed control room and equipment operator response to the adverse environmental condition, including additional monitoring of affected plant equipment, such as emergency core cooling system pumps and heat exchangers, to ensure the systems remained capable of performing their safety functions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown (3 samples)

a. Inspection Scope

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

- Salem Unit 3 gas turbine generator starting air supply system and fuel oil supply system with one starting air compressor out of service and alternate means of pressurizing the starting air system staged for contingency use during severe hot weather;
- Salem Unit 2 residual heat removal (RHR) system following preventive maintenance activities inside the RHR equipment rooms; and
- Salem Units 1 and 2 service water as service water temperature approached technical specification limits.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown (71111.04S - 1 sample)

a. Inspection Scope

The inspectors conducted one complete walkdown of accessible portions of the Salem Unit 1 auxiliary building control air system to verify that the system was properly configured, hangers and supports were correctly installed and functional, compressor oil reservoir levels were normal, and to identify any discrepancies between the existing lineup and the prescribed lineup. The inspectors interviewed the system engineer and reviewed corrective action evaluations associated with the system to determine whether equipment alignment problems were being identified and corrected at a low threshold. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Protection - Tours

a. Inspection Scope (9 samples)

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

- Unit 1 and Unit 2 auxiliary feedwater pumps area;
- Unit 1 and Unit 2 emergency diesel generator (EDG) area;
- Unit 1 and Unit 2 volume control and boric acid tanks area;
- Unit 1 and Unit 2 electrical penetration area; and
- Unit 1 auxiliary building: 55 foot elevation and the number 11 residual heat removal pump and heat exchanger room (45 foot elevation).

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation (71111.05A)

a. Inspection Scope (1 sample)

The inspectors observed one unannounced fire drill conducted in the fresh water pump house on August 2, 2006. The drill was observed to evaluate the readiness of the plant fire brigade to fight fires. The inspectors verified that PSEG staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient fire fighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope (1 sample)

External Flooding Review. The inspectors performed one external flood protection measures inspection for Salem Units 1 and 2. Numerous watertight flood protection doors, auxiliary building penetration seals credited for wave runup protection, and the service water intake structure were walked down to verify operational readiness. The inspectors assessed the readiness of portable sump pumps and interviewed the external flood protection engineer. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope (1 sample)

Resident Inspector Quarterly Review. The inspectors observed a simulator training scenario conducted on August 22, 2006, to assess operator performance and training effectiveness. The scenario involved a reactor trip and steam generator tube rupture. The inspectors verified operator actions were consistent with operating, alarm response, abnormal, and emergency procedures. The inspectors assessed simulator fidelity and verified that evaluators identified deficient operator performance where appropriate. The inspectors observed the simulator instructors' critique of operator performance. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (2 samples)

The inspectors reviewed performance monitoring and maintenance effectiveness issues for the Salem gas turbine generator and 1C EDG unavailability documented in PSEG notification 20285254. The inspectors assessed whether PSEG was adequately monitoring equipment performance to ensure that preventive maintenance was effective. The inspectors verified that the components were monitored in accordance with the maintenance rule (MR) program requirements. The inspectors compared documented functional failure determinations and unavailability hours to those being tracked by PSEG to evaluate the effectiveness of PSEG's condition monitoring activities and to determine whether performance goals were being met. The inspectors reviewed applicable work orders, corrective action notifications, preventive maintenance tasks, and system health reports. The inspectors also interviewed responsible system engineers. Documents reviewed are listed in the attachment.

b. Findings

(Closed) URI 05000311/2006003-01, 22 Service Water Strainer Multiple Trips

Introduction: The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to accomplish maintenance in accordance with procedures when the 22 service water strainer (SWS) tripped on March 10 and 11, 2006.

Description: On March 3, 2006, PSEG personnel inspected and adjusted the 22 SWS under preventive maintenance order 30083454. Between March 3 and March 10, the 22 service water pump (SWP) and 22 SWS were operated routinely for an accumulated time of 114 hours. On March 10, 2006, the 22 SWS motor breaker tripped open. The operators reset the thermal overload protective devices for the SWS motor, which PSEG procedures allow to be performed once without determining the cause of the trip. Operators then tested rotation of the 22 SWS without running the 22 SWP. The 22 SWP and associated SWS were next started on March 11, 2006, and operated for approximately two hours before the 22 SWS motor breaker tripped again.

PSEG entered this issue into the corrective action program as notification 20275035 and completed corrective maintenance under notification 20275118. In response to NRC inspector questions regarding the cause, PSEG determined the 22 SWS tripped because internal drum-to-body clearances were too small following the maintenance on March 3. The drum-to-body clearance is the distance between the inner rotating basket and the adjacent tapered stationary body.

The inspectors determined that PSEG procedure SC.MD-PM.SW-0003, "Service Water Auto Strainer Adjustment, Inspection, Repair and Replacement," called for a drum-to-body clearance of 0.010 to 0.063 inches. However, in response to inspector questions, PSEG found that vendor documentation specified a minimum drum-to-body clearance of 0.045 inches to provide sufficient clearance for an internal seal ring. Smaller drum-to-body clearances could cause increased resistance to strainer rotation because river water detritus or the seal ring itself could be pinched between the drum and body. PSEG initiated corrections to the applicable maintenance procedure (20280934).

The inspectors' further review of maintenance documents for March 3, 2006, determined that maintenance personnel adjusted the drum-to-body clearances during the preventive maintenance activity, based on the documented as-found and as-left drum-to-body clearances. The inspectors noted that section 5.3 of procedure SC.MD-PM.SW-0003 directed adjustment of the drum clearances and tightening of the lower locknut. However, the technicians marked section 5.3 of procedure SC.MD-PM.SW-0003 as "not applicable," even though they adjusted the drum clearances. Finally, the inspectors determined that the drum-to-body clearance decreased while 22 SWS operated between March 3 and March 10, 2006, because the recorded as-left gap on March 3, 2006, ranged from 0.035 to 0.055 inches, and the as-found gap on March 11, 2006, was approximately 0.020 inches. Following corrective maintenance, on March 12, 2006, the drum-to-body clearances were approximately 0.050 to 0.063 inches.

PSEG re-evaluated the 22 SWS trip and determined the 22 SWS internal clearances decreased during operation (70057833). PSEG identified two possible causes for the decrease. First, the 22 SWS lower locknut may not have been tightened. Second, the 22 SWS lower locknut may not have been locked in place. Either cause would have decreased the 22 SWS internal clearances by allowing the internal rotating drum to drift lower into the tapered external stationary body. The inspectors concluded both locknut tightening and locknut locking were required by SC.MD-PM.SW-0003, section 5.3.

Analysis: The inspectors determined that not completing the procedure steps to tighten or lock the 22 SWS lower locknut was a performance deficiency that caused 44 hours of 22 SWP unavailability following the first trip on March 10, 2006. The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone, and it affected the cornerstone objective to limit the likelihood of those events that could upset plant stability and challenge critical safety functions during power operations. Unavailability of the 22 SWS and SWP increased the likelihood of a loss of service water. This finding also impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, incorrectly performed maintenance degraded both availability and reliability of the 22 SWS and SWP. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a Phase 1 SDP screening and determined that a more detailed Phase 2 evaluation was required to assess the safety significance because the performance deficiency affected two cornerstones. However, the Risk-Informed Inspection Notebook for Salem Nuclear Generating Station does not evaluate loss of service water initiating events. Therefore, an NRC Region 1 Senior Reactor Analyst (SRA) conducted a Phase 3 analysis.

The SRA’s Phase 3 analysis determined that the finding was of very low safety significance (Green). The analysis used the NRC’s Standardized Plant Analysis Risk (SPAR) model, Revision 3.22, for the Salem facility, modified for all high temperature reactor coolant pump (RCP) seals installed at Unit 2, and assumed the 22 SWP was out-of-service for 44 hours and that the loss of service water initiating event frequency increased during this time because of lost redundancy in the service water trains as a result of the performance deficiency. The increase in core damage frequency due to internally initiated events was in the low E-8 range (an increase in the core damage frequency in the range of 1 core damage accident in 30,000,000 years of reactor operation). The dominant accident sequence involved a loss of service water initiating event, assuming no recovery of service water. Core damage then results following a reactor coolant pump seal failure due to lack of cooling and the failure of high pressure recirculation.

This finding had a cross-cutting aspect in the area of human performance related to the work practices component, because PSEG did not effectively communicate expectations regarding procedure compliance and personnel did not follow procedures. Consequently, maintenance technicians did not adhere to established maintenance procedures and caused 44 hours of 22 SWS unavailability.

Enforcement: 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Section 5.3 of SC.MD-PM.SW-0003, “Service Water Auto Strainer Adjustment, Inspection, Repair and Replacement,” Revision 25, directs adjustment of the drum clearances. Step 5.3.5 directs tightening of the lower locknut, and steps 5.3.6 and 5.3.7 direct installation of the lower lockplate. Contrary to the above, PSEG did not accomplish the 22 SWS inspection and adjustment on March 3, 2006, in accordance with prescribed procedures. Specifically, section 5.3 of SC.MD-PM.SW-0003 completed on March 3 was marked "not applicable." As a result, the 22 SWP was rendered inoperable for 44 hours, starting on March 10, 2006. Because this finding is of very low safety significance and has been entered into the corrective action program in notifications 20275035 and 20286677, this violation is being treated as a NCV, consistent with section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2006004-01, 22 Service Water Strainer Trip)**

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

a. Inspection Scope (6 samples)

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

- Number 1 station air compressor (SAC), number 3 SAC, and Salem Generating Station Unit 3 (gas turbine generator) on July 10, 2006;
- the emergent unavailability of the 1B vital instrument bus inverter concurrent with emergent unavailability of the Hope Creek 5015 Red Lion 500 kV transmission line on July 24, 2006;
- the emergent unavailability of the 22 service water pump concurrent with number 3 SAC out of service on July 26, 2006;

- the emergent unavailability of the1C emergency diesel generator (EDG) pre-lube oil heater on July 27 and 28, 2006;
- 13 service water strainer outage concurrent with number 3 SAC out of service on August 22, 2006; and
- concurrent outages of the 11 diesel fuel oil transfer pump, 16 service water pump, and number 3 station air compressor on September 14, 2006.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (5 samples)

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

- Notification 20290419, operability of the 23 component cooling pump with an oil leak from the outboard pump bearing;
- Notification 20288276, potential overloading of an emergency AC bus during a postulated accident caused by loss of control power to the 12 spent fuel pool cooling pump;
- Notification 20293612, pressurizer level channel 1B affected by containment temperature;
- Notification 20292480, elevated service water temperature; and
- Notification 20276469, operability of the 1C emergency diesel generator following an overcrank condition.

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

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (5 samples)

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

- Work orders (WO) 30129614, 30132803 and 30138293, 11 control air dryer preventive maintenance activities during replacement of 3 station air compressor;
- WO 60064179, 22 service water pump and strainer corrective maintenance;
- WO 60062345, 13 chemical and volume control reciprocating charging pump corrective maintenance;
- WOs 30126596, 30120074, 50098442, and 50097115, 11 diesel fuel oil storage tank transfer pump preventive maintenance; and
- WO 60064079, 1B vital instrument bus inverter corrective maintenance.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope (1 sample)

On September 26 and 27, 2006, the inspectors reviewed the Unit 2 forced outage work scope associated with a manual reactor trip on September 26, 2006. The inspectors appropriately considered plant shutdown risk and maintained defense-in-depth systems while Unit 2 remained in hot standby conditions. The inspectors reviewed PSEG's post reactor trip review and apparent cause reports, and observed reactor startup to criticality. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (6 samples)

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

- WO 50084209, moderator temperature coefficient measurement for Unit 2;
- WO 50095546, 2PT-505 turbine steam line inlet pressure functional test;
- WO 50096094, 12 safety injection pump inservice testing;
- WO 50096590, 22 auxiliary feedwater pump inservice testing;
- WO 50094784, local leak rate testing of containment isolation valves 1VC1 and 1VC2 (containment atmosphere ventilation purge supply line); and

- WO 50095714, 2A emergency diesel generator surveillance.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

The inspectors reviewed the installation of four temporary air compressors as a temporary modification and assessed whether PSEG followed its administrative process for implementing the temporary modification. The temporary air compressors were installed as a compensatory measure while PSEG personnel replaced the number 3 station air compressor (SAC). The inspectors compared the associated 10 CFR 50.59 screenings for use of the temporary air compressor and for replacement of the number 3 SAC against the UFSAR system description to determine that the design basis was maintained. The inspectors walked down the temporary air compressor installation and observed the operation of the compressors to verify functionality. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors evaluated an emergency preparedness drill from the control room simulator during a licensed operator requalification examination on August 1, 2006. The inspectors evaluated drill performance relative to developing proper classifications, notifications, and protective action recommendations by PSEG personnel. The inspectors reviewed the Salem Event Classification Guides and Emergency Plans to determine classifications and notifications were in accordance with these documents. The inspectors also verified that PSEG correctly counted the drill contribution in the NRC performance indicator for Drill/Exercise Performance.

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES [OA]**

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the Corrective Action Program

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

.2 Annual Sample: Reactor Coolant Pump Seals

a. Inspection Scope

The inspectors reviewed PSEG's evaluation of Westinghouse Technical Bulletin TB-04-22, "Reactor Coolant Pump (RCP) Seal Performance and Appendix R Compliance," Rev. 0 and 1, and PSEG's evaluation of NRC Information Notice (IN) 2005-14, "Fire Protection Findings on Loss of Seal Cooling to Westinghouse RCPs." If RCP seal cooling is postulated to be lost during an accident or transient, the RCP seals can heat up, resulting in potentially increased leakage. These documents identify that if operators reestablish seal cooling after the seal temperature exceeds approximately 235 degrees F, the cooling water could thermally shock the RCP seals, resulting in leak rates greater than the charging system capacity. The inspectors reviewed various notifications and work orders, and conducted interviews with plant staff. The inspectors evaluated PSEG's actions documented in the corrective action program. The inspectors also reviewed operating procedures to ensure the procedures were sufficient to prevent RCP seal damage under the above conditions.

b. Findings and Observations

No findings of significance were identified.

The inspectors concluded that PSEG properly implemented guidance to ensure seal leakage during a postulated transient would not exceed the flow rate of one charging pump and that thermal shock to the RCP seals would not occur. The inspectors identified two observations regarding PSEG's evaluation of this issue.

The inspectors identified that preventative maintenance was not completed on several RCP seal temperature indicators on both units. In response, PSEG provided acceptable interim guidance to reactor operators, and subsequently calibrated the instruments. The inspectors determined that this issue was minor because although several RCP seal temperature indicators were out of calibration, there was no impact on operability or reliability, due to the availability of accurate, redundant instrumentation.

The inspectors also observed that PSEG did not explicitly evaluate the Westinghouse recommendation to cool the RCP seals down at less than 1 degrees F per minute with temperatures above 235 degrees F to prevent RCP shaft warping during a postulated loss of RCP seal cooling event. However, this observation was minor because existing PSEG procedures used natural circulation to cooldown at a rate less than the Westinghouse guidance.

.3 Safety Conscious Work Environment Metric Review

a. Inspection Scope

The inspectors reviewed PSEG's progress in addressing safety conscious work environment (SCWE) issues that were discussed in the NRC's annual assessment letter dated March 3, 2006. In that letter, the NRC staff documented a SCWE substantive cross-cutting issue and stated the NRC would to continue to monitor progress in this area.

The inspectors completed a sampling review of PSEG's SCWE metrics, or performance indicators (PIs), for second quarter 2006. Documents reviewed are listed in the attachment.

b. Findings and Observations

No findings of significance were identified.

In the second calendar quarter of 2006, PSEG identified twenty-four PIs as being green or satisfactory while five PIs were identified as red or needing improvement. This was consistent with the first quarter 2006, when there were twenty-four green PIs and six red PIs. One of the green performance indicators reported in the first quarter 2006 documented the results of a Synergy Consulting Services Corporation survey of the workforce completed in the first quarter of 2006. This PI was not included in the second quarter results.

4OA3 Event Followup (71153 - 2 samples)

a. Inspection Scope

.1 1B Vital Instrument Bus Inverter

a. Inspection Scope

A lightning storm on July 22, 2006, caused the Hope Creek Generating Station 5015 Red Lion 500 kV transmission line (5015 line) to de-energize. Consistent with procedures and prearranged contingency actions, Unit 1 operators initiated a power reduction to 40% of rated core thermal power. The 5015 line was returned to service before the power reduction was completed. Therefore, operators stabilized Unit 1 at 54% power. The initiating electrical transient also indirectly caused the 1B vital instrument bus (VIB) inverter to trip. PSEG repaired the 1B VIB inverter and returned it to service. Operators returned Unit 1 to 100% power on the same day.

The inspectors reviewed the operating crew performance for conformance to technical specification limiting conditions for operation and adherence to plant operating procedures. The inspectors also observed the Unit 1 return to full power. The inspectors observed corrective maintenance and post-maintenance testing of the 1B VIB inverter. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Unit 2 Manual Reactor Trip

a. Inspection Scope

On September 26, 2006, operators manually tripped Unit 2 when the 21 reactor coolant pump seal return flow rate exceeded six gallons per minute. The reactor trip was required by Salem procedure S2.OP-AB-RCP-0001, “Reactor Coolant Pump Abnormality,” which directs operators to trip the reactor, stop affected reactor coolant pumps, and isolate seal return flow, if seal return flow exceeds six gallons per minute. PSEG determined the excessive seal return flow rates were temporary. The transient caused elevated seal return flow rates from all four reactor coolant pump seals. The 21 reactor coolant pump seal was the only reactor coolant pump to exceed six gallons per minute. Shortly after the trip, the 22, 23, and 24 seal return flow rates returned to normal, and there was no evidence of damage to the reactor coolant pump seals. Consequently, operators restored seal return flow from 21 reactor coolant pump, observed that the seal return flow rate returned to normal, and returned 21 reactor coolant pump to service. PSEG returned Unit 2 to power operation on September 27, 2006. PSEG is conducting a root cause evaluation to confirm the causes of this event.

The inspectors observed control room operator performance as the operating crew completed emergency operating procedures, abnormal operating procedures and standard operating procedures to establish steady state operation in Mode 3, “Hot Standby.” The inspectors reviewed PSEG’s prompt investigation report and adverse condition monitoring plan to closely monitor RCP seal performance during plant restart. The inspectors reviewed numerous plant parameters to validate PSEG’s conclusions that the likely cause was associated with end-of-cycle reactor coolant system chemical concentrations.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 05000311/2005003-03, Assessment of Reactor Coolant System Instrument Tube Structural Integrity

Introduction: The inspectors identified a Green non-cited violation because PSEG did not follow Salem Technical Specification 3.4.11.1.b., Structural Integrity. PSEG discovered a leak on the instrument tubing for reactor coolant system loop flow transmitter 2FT416 and did not properly classify and evaluate the leak for operability or structural integrity, or alternatively isolate the affected tubing.

Description: On April 4, 2005, during a containment walkdown with the plant at 100% power prior to the 2R14 refueling outage, PSEG identified a through-wall leak on the instrument tubing for RCS flow transmitter 2FT416. The affected instrument tubing is ASME Code Class 2, stainless steel instrument tubing. The through-wall leak called into question the structural integrity of this tubing.

Technical Specification (TS) 3.4.11.1.b specifies, in part, "With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements (i.e., structural integrity), restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200 degrees F."

PSEG procedure SH.OP-AP.ZZ-0108(Q), "Operability Assessment And Equipment Control Program," Revision 19, provides guidance to operators in implementing TS 3.4.11.1.b. Attachment 1, Paragraph E of this procedure states, in part: "TS do not allow for pressure boundary leakage. The TS define the Pressure Boundary Leakage as leakage through a non-isolable fault in the reactor coolant system component body, pipe wall or vessel wall. Upon discovery of leakage from a Class 1, 2, or 3 component pressure boundary (e.g., pipe wall, valve body or pump casing) the component is not operable." Further, the procedure provides guidance relative to the need for flaw characterization and evaluation to justify continued operation. However, PSEG did not perform and document a flaw characterization or evaluation to demonstrate structural integrity per TS 3.4.11.1.b and justify continued operation with this through-wall leak.

PSEG initiated notification 20231322 to document the leak. As documented in notification 20231322, PSEG determined that it was impractical to isolate the leaking tubing, considering the potential for perturbing the flow transmitter. However, the inspectors noted the associated isolation valve was available and the leaking tubing was isolable. Though Technical Specification 3.4.11.1.b required PSEG to restore structural integrity or isolate the affected component, the inspectors determined neither action was accomplished. The plant continued to operate at 100% power until the 2R14 refueling outage began on April 5, 2005.

Analysis: PSEG did not meet the requirements of Salem Technical Specification 3.4.11.1.b. when operation was continued with a through-wall leak in 2FT416 flow transmitter tubing. This condition existed from the morning of April 4, 2005, until the afternoon of April 5, 2005, a period of approximately 36 hours. This issue was a performance deficiency because PSEG did not meet the requirements of the Technical

Specification and the guidance of procedure SH.OP-AP.ZZ-0108(Q), and the condition was reasonably within PSEG's ability to foresee and correct.

This finding is more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that could upset plant stability and challenge critical safety functions during power operations. The tubing leak affected the barrier integrity example within the equipment performance attribute.

The inspectors determined that the finding was of very low safety significance (Green) using a Phase 1 screening in Appendix A of Inspection Manual Chapter 0609, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." It is expected that a tubing crack would result in an increase in RCS leakage, and operators would take action prior to exceeding Technical Specification limits for RCS leakage. Therefore, assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function.

This performance deficiency has a cross-cutting aspect in the area of problem identification and resolution, in the corrective action program component, because PSEG did not throughly evaluate the condition. This includes properly classifying, prioritizing, and evaluating for operability conditions adverse to quality. The tubing leak was not properly classified as through-wall and evaluated for operability and structural integrity.

Enforcement: Technical Specification 3.4.11.1.b states, in part, "With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the RCS temperature above 200 degrees F. Contrary to the above, on April 4, 2005, through April 5, 2005, PSEG did not follow the required actions to restore structural integrity or isolate a leaking RCS component with a through-wall leak when a leak was discovered on the instrument tubing for loop flow transmitter 2FT416. Because this finding is of very low safety significance and has been entered into the corrective action program as notification 20297120, this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000311/2006004-02, Reactor Coolant System Tubing Structural Integrity)**

4OA5 Other

.1 (Open) URI 05000272/2003006-02, NRC to Review Results of Unit 1 Spent Fuel Pool Structural Integrity Analysis

During a review of through-wall leakage of spent fuel pool water from PSEG's Salem Unit 1 spent fuel pool, NRC inspectors questioned potential structural impacts associated with the leakage. PSEG reviewed this matter, did not identify any immediate safety concerns associated with the leakage, and did not identify evidence that the structure had been adversely affected such that it would not meet its design function. Notwithstanding, PSEG initiated an evaluation of potential long term impacts on the structure. As part of that evaluation, PSEG initiated extensive laboratory testing of representative samples of concrete and reinforcing steel samples to evaluate potential impacts of the borated water on the structure. PSEG conducted visual inspections of the structure and evaluated industry experience on borated water impact on structures. The NRC's Office of Nuclear Reactor Regulation (NRR) has been assisting NRC Region I in reviewing PSEG's activities in this matter to resolve remaining questions. In February 2006 representatives of NRR and Region I met with PSEG and its technical contractor, toured the affected areas, and reviewed assessments of the integrity of the spent fuel pool structure. Following that meeting, NRR personnel reviewed additional materials and information and informed Region I of several remaining questions that should be addressed in order to establish a conclusion on this matter. NRR and Region I will provide the remaining questions to PSEG for resolution. This URI remains open pending resolution of these questions by PSEG and completion of reviews by NRC personnel.

.2 (Open) 05000272&311/2005002-03, Ground Water Intrusion to the Auxiliary Building and Containment Building Seismic Gap

The inspectors reviewed an issue related to the potential long term impacts on concrete and reinforcing bar in the auxiliary building and containment building seismic gap areas due to boric acid in ground water, which is related to URI 05000272/2003006-02, NRC to Review Results of Unit 1 Spent Fuel Pool Structural Integrity Analysis. NRR and Region I will provide additional questions to PSEG with the set of questions for the Unit 1 spent fuel pool structural integrity analysis, as discussed above. This URI remains open pending resolution of these questions by PSEG and completion of reviews by NRC personnel.

4OA6 Meetings, Including Exit

On October 5, 2006, the resident inspectors presented the inspection results to Mr. Carl Fricker, Salem Plant Manager, and other members of PSEG staff. None of the information reviewed by the inspectors was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

T. P. Joyce, Salem Vice President
C. J. Fricker, Salem Plant Manager
S. J. Robitski, Salem Engineering Director
T. Gierich, Salem Operations Manager
J. Stone, Salem Maintenance Director
G. J. Sosson, Salem System Engineering Manager
A. T. Roberts, Manager - Engineering Programs
R. S. Gary, Salem Technical Superintendent - Radiation Protection
S. R. Mannon, Salem Regulatory Assurance Manager
W. Treston, PSEG ISI Manager
T. Roberts, Materials Manager
C. Pupek, PRA Engineer
E. H. Villar, Nuclear Licensing/Compliance
J. M. Wearne, Nuclear Licensing/Compliance
T. R. Wathey, Salem Reactor Engineering Manager
K. J. Weigel, Salem NSSS Engineering Supervisor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened/Closed

05000311/2006004-01	NCV	22 Service Water Strainer Trip (Section 1R12)
05000311/2006004-02	NCV	Reactor Coolant System Tubing Structural Integrity (Section 4OA3)

Closed

05000311/2006003-01	URI	22 Service Water Strainer Multiple Trips (Section 1R12)
05000311/2005003-03	URI	Assessment of Reactor Coolant System Instrument Tube Structural Integrity (Section 4OA3)

Discussed

05000272/2003006-02	URI	Unit 1 Spent Fuel Pool Structural Integrity Analysis (Section 4OA5)
05000272/2005002-03 and 05000311/2005002-03	URI	Ground Water Intrusion to the Auxiliary Building and Containment Building Seismic Gap (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

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

Section 1R01: Adverse Weather Protection

Procedures
SC.OP-AB.ZZ-0003, Component Fouling, Rev. 10
SC.OP-AB.ZZ-0001, Adverse Environmental Conditions, Rev. 9

Section 1R04: Equipment Alignment

Procedures
S3.MD-SP.JET-0001, Pressurization of S3 Airpack by Alternate Means, Rev. 1
S3.OP-SO.JET-0001, Gas Turbine Operation, Rev. 23
S1.OP-AB.CA-0001, Loss of Control Air, Rev. 15

Drawings
205249, 252945, 205332, 229988, 205243, 205217, 205200,

Notifications
20290718, 20284078, 20290724, 20290892, 20290967, 20291034, 20291377, 20291414, 20291769, 20291440, 20291467, 20295752, 20295753, 20295754, 20295755, 20295756, 20295757, 20295758, 20295759, 20295760, 20295771, 20295245

Orders
60064034, 60063170,

Other Documents
Salem Inservice Testing Program Basis Data Sheets - Valves,
Control Air System Lineup 740 printed from SAP on July 13, 2006

Section 1R05: Fire Protection

Procedures
FRS-II-433, Pre-Fire Plan U1 & U2 Auxiliary Feed Water Pumps Area, Rev. 5
FRS-II-445, Pre-Fire Plan U1 & U2 Diesel Generator Area, Rev. 10

SC.FP-AP.ZZ-0003, Actions for Inoperable Fire Protection - Salem Station, Rev. 11
FRS-III-815, Pre-Fire Plan Fire/Fresh Water Pump House, Rev. 1
FRS-II-411, Pre-Fire Plan U1 & U2 Reactor Plant auxiliary Equipment Area Elevations: 45' &55', Rev. 2
FRS-II-454, Pre-Fire Plan U1 & U2 Volume Control & Boric Acid Tanks Area, Rev. 2
FRS-II-511, Pre-Fire Plan U1 & U2 Electrical Penetration Area, Rev. 4
S1.FP-ST.FBR-0028(Q), Class 1 Fire Damper Operability Test, Rev. 4
S1.FP-SV.FBR-0031(Q), Class 1 Fire Damper Visual Inspection, Rev. 3
S2.FP-ST.FBR-0028(Q), Class 1 Fire Damper Operability Test, Rev. 4
S2.FP-SV.FBR-0031(Q), Class 1 Fire Damper Visual Inspection, Rev. 4

Drawings
207627, Aux. Bldg. El. 122' Ventilation Ducts

Notifications
20109598, 20221470, 20277361, 20294855

Other Documents
Salem and Hope Creek Fire Impairment Log Book, dated July 27, 2006
Fire Drill Scenario S4UA080206 SAP ID-51220263
Generic Letter 86-10, Implementation of Fire Protection Requirements
Salem Individual Plant Examination for External Events, Section 4, Internal Fire Analysis

Section 1R06: Flood Protection Measures

Procedures
NC.OP-DG.ZZ-0002, Severe Weather Guide, Rev. 6
S1.OP-DL.ZZ-0003(Q), Control Room Log - Modes 1-4, Rev. 50
SC.MD-PM.ZZ-0036(Q), Watertight Door Inspection and Repair, Rev. 5
SC.OP-AB.ZZ-0001(Q), Adverse Environmental Conditions, Rev. 9

Notifications
20050752, 20159188, 20202390, 20246979, 20247077, 20251172, 20272437, 20283384

Orders
60056620, 70013154, 70050139, 70057254, 80084022

Other Documents
Salem Individual Plant Examination for External Events, Section 5, High Winds, Floods and Other External Events
Technical Specifications 3/4.7.5, Flood Protection
Updated Final Safety Analysis Report, Figure 2.4-2, Yard Drainage System
Updated Final Safety Analysis Report, Figure 2.4-3, Service Water Intake
Updated Final Safety Analysis Report, Section 2.4, Hydrologic Engineering
Updated Final Safety Analysis Report, Section 3.4, Water Level (Flood) Design

Section 1R11: Licensed Operator Requalification Program

Other Documents
Simulator Training Scenario, AB.ROD-3, AP.RCP-1, SGTR1 and 3, S-RSG-068

Section 1R12: Maintenance Effectiveness

Notifications
20251471, 20285254, 20285636, 20285638, 20285789, 20251990, 20251698, 20272873, 20272964, 20294630

Orders
70049683, 70057799, 60062825, 70057896, 70050150, 60062984, 70038389

Other Documents
Adverse Condition Monitoring plan 06-017, Emergency Diesel Generator Jacket Water Expansion Tank Level Monitoring.
Operability Determination 06-013, Reliability of Salem Unit 1 and Unit 2 Emergency Diesel Generators with Potentially Degraded Heads Supplied by Canadian Allied Diesel
1C Emergency Diesel Generator Oil Sample Analysis Report of crankcase oil sample drawn on May 8, 2006
Troubleshooting Log of 1C Emergency Diesel Generator under notification 20285254
Salem 1C Emergency Diesel Generator Jacket Water Leak Interim Findings Report dated May 24, 2006.
ALCO Vendor Technical Manual 301103
Root Cause Investigation Report re: Jacket Water Leak on Salem Unit 1 "C" Diesel 5R Cylinder MPR Associates, Inc., ALCO 251 EDG Cylinder Head Leak (root cause of a leak which occurred on the 5L cylinder head on August 30, 2005)
Artech Testing, LLC, Failure Analysis of ALCO Diesel Cylinder Head (analysis of the 5L cylinder head which leaked on August 30, 2005)
PSE-06504, Exelon Power Labs, Salem 1C EDG 5R Cylinder Head Pressure Testing and Failure Evaluation of the Fuel Injector Sleeve, Salem Unit 1, dated June 29, 2006
NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 4
SH.ER-DG.ZZ-0002(Z), Form-1 (A)(1) Goal Action Plan, Unit 3 Gas Turbine System

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
SH.OP-AP.ZZ-0027, On-Line Risk Assessment, Rev. 12
S2.OP-SO.SW-0001, Service Water Pump Operation, Rev. 21
S2.OP-SO.SW-0005, Service Water System Operation, Rev. 34
S1.OP-SO.DG-0003, 1C Diesel Generator Operation, Rev. 35
Salem FSAR Section 8.3.1.4, Rev. 18
SC.OP-SO.500-0001, Trip-a-Unit Scheme Operation, Rev. 4
S1.OP-SO.115-0012, 1B Vital Instrument Bus UPS System Operation, Rev. 12
S1.OP-SO.115-0005, 11 Emergency Lighting UPS System Operation, Rev. 8

SC.MD-PM.SW-0003(Q), Service Water Auto Strainer Adjustment, Inspection, Repair and Replacement, Rev. 25

Drawings
211370, 309945, 203007, 211365, 20294580, 20294645, 20294739

Notifications
20287888, 20292024, 20292066, 20292102, 20291480, 20291578, 20291577, 20292232, 20292645

Orders
60064079, 30136683, 60052088, 70058966, 70060333

Other Documents
NRC Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants, dated May 2000
NUMARC 93-01, Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Rev. 2
Salem Generating Station (SGS) PRA Risk Evaluation Form for Work Week Nos. 132, 134, and 141 (weeks of July 9, 2006, July 23, 2006, & September 10, 2006)
SH.OP-DL.ZZ-0027 Form 5, 1C EDG Lube Oil Sump Temperature Operator Action Log, dated July 27-28, 2006
Work Clearance Documents 4184922 & 4184929

Section 1R15: Operability Evaluations

Procedures
S1.OP-ST.SSP-0003(Q), SEC Mode OP Testing 1B Vital Bus, Rev. 21

Drawings
601232, 1B-460V Vital Bus One-Line, Rev. 15

Notifications
20276759, 20276469, 20290419, 20291260, 20291279, 20288276, 20292480, 20293612

Orders
70058287, 70057896, 70059869

Other Documents
Trico Manufacturing Corp., Technical Information Sheets Glass, LS, or SS Opto-Matic Oilers, Calculation S-C-ZZ-MDC-1947, Post-LOCA Vital Area Access Mission Doses - AST, Rev. 0,
Calculation S-C-VAR-MDC-1518, Postaccident Access to Vital Areas, Rev. 4,
Calculation S-C-SJ-MEE-1978, Required Mission Times for Salem ECCS Pumps During Recirculation Phase, Rev. 0,
Salem Unit 1 Control Room Narrative Log, June 4, 2006
Updated Final Safety Analysis Report, Section 8.3, Onsite Power System

CROD 06-018 (70059869), Elevated Service Water Temperatures
CROD 06-019 (70060381), Salem Unit 1 Operability Determination
SER OTDM 06-012, Issue Resolution Documentation form, NC.CA-DG.ZZ-0102
Adverse Condition Monitoring and Contingency Plan 06-010
No. 1C EDG Trip on Overcrank (80088928/0010)
Jacket Water Leak on Salem Unit 1 C Diesel 5R Cylinder (70057896)

Section 1R19: Post-Maintenance Testing

Procedures
NC.NA-AP.ZZ-0050, Station Post Maintenance Testing, Rev. 7
NC.MD-AP.ZZ-0050, Maintenance Testing Program Matrix, Rev. 7
SH.MD-GP.ZZ-0240, System Pressure Test at Normal Operating Pressure and Temperature, Rev. 7
SC.MD-CM.CVC-0001, Numbers 13 and 23 Charging Pump Repacking Plunger and Valve Repair or Replacement, Rev. 10
S1.OP-ST.CVC-0005, Inservice Testing - 13 Charging Pump, Rev. 15
S1.RA-ST.CVC-0005, Inservice Testing 13 Charging Pump Acceptance Criteria, Rev. 9
S1.OP-ST.DG-0004, 11 Fuel Oil Transfer System Operability Test, Rev. 20
S1.RA-ST.DG-0004, Diesel Generator Auxiliaries 1 Fuel Oil Transfer System Operability Test Acceptance Criteria, Rev. 7
SC.MD-PM.ZZ-0018, AC Motor Cleaning and Inspection, Rev. 7
SC.MD-PM.DG-0006, Diesel Generator Fuel Oil Transfer and Prelubrication Oil Pump Internal Inspection and Seal Replacement, Rev. 7
SH.MD-GP.ZZ-0022, Bolt Torquing and Bolting Sequence Guidelines, Rev. 1
SC.MD-PM.115-0001, 10/12 KVA Vital Instrument Bus Inverter Preventive Maintenance, Rev. 10
SC.MD-CM.115-001, 10/12KVA Uninterruptible Power Supply Troubleshooting and Repair, Rev. 9

Notifications
20291242, 20291480, 20292024, 20294644, 20295304, 20295306, 20295308, 20297184

Orders
30129614, 30132803, 30138293, 60064179, 30126596, 30120074, 50098442, 50097115, 60064079

Other Documents
MA-AA-716-004, Conduct of Troubleshooting, Rev. 5

Section 1R20: Refueling and Other Outage Activities

Procedures
S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load, Rev. 26
S2.OP-IO.ZZ-0004, Power Operation, Rev. 55

Section 1R22: Surveillance Testing

Procedures
SC.RE-ST.ZZ-0007, Moderator Temperature Coefficient Measurement, Rev. 9
S2.RE-RA.ZZ-0012, Salem Generating Station/Reactor Engineering Figures, Rev. 95
S2.IC-FT.RCP-0024, 2PT-505 Turbine Steam Line Inlet Pressure Protection Channel I, Rev. 9
S1.OP-ST.SJ-0002, Inservice Testing - 12 Safety Injection Pump, Rev. 15
S1.OP-ST.SJ-0002, Inservice Testing 12 SI Pump Surveillance Data Acceptance Criteria, Rev. 5
SC.RA-AP.ZZ-0051, Leakage Monitoring and Reduction Program, Rev. 1
S2.OP-ST.AF-0002(Q), Inservice Testing - 22 Auxiliary Feedwater Pump, Rev. 16
S2.OP-ST.DG-001(Q), 2A Diesel Generator Surveillance Test, Rev. 44
S2.RA-ST.AF-0002(Q), Inservice Testing 22 Auxiliary Feedwater Pump Acceptance Criteria, Rev. 7
S1.OP-LR.VC-0001, Type C Leak Rate Test 1VC1 and 1VC2, Rev. 0
SH.RA-DG.ZZ-0113, Qualification of Leak Rate Monitor Technicians Desk Top Guide, Rev. 0
ER-AA-380, Primary Containment Leakrate Testing Program, Rev 4

Drawings
205234, 205336, 205238

Notifications
20282082, 20265800, 20266402, 20268702, 20290614, 20284007, 20287220, 20292525, 20190639, 20202952, 20226617

Orders
60062495, 60056753, 60060070, 70052624, 70053241, 50097289, 50097398, 50095546, 50096094, 70057489, 60048253, 70045294, 50094784, 50095714

Other Documents
Nuclear Design and Startup Report, Salem Unit 2, Cycle 15 (NFS-0244), Rev. 1
Memorandum from Thomas R. Wathey to Shift Managers - Salem dated June 30, 2006 (SRE 06-001), Subject: Salem Unit 2 Cycle 15 Moderator Temperature Coefficient (MTC) Measurement
NRC Inspection Procedure 61708, Isothermal and Moderator Temperature Coefficient Determinations, Issue Date October 11, 1985
Salem Inservice Testing Program Basis Data Sheets - 22 Auxiliary Feedwater Pump, Table 8-2B, Rev. 8
Technical Specifications 4.7.1.2.b.1, Auxiliary Feedwater System
Updated Final Safety Analysis Report, Section 10.4.7.2, Auxiliary Feedwater System

Section 1R23: Temporary Plant Modifications

Procedures
S1.OP-SO.CA-0001, Control Air System Operation, Rev. 12
SC.OP-SO.SA-0002, Operating Temporary Air Compressors - Station Air, Rev. 0
SC.OP-SO.SA-0002, Temporary Station Air Compressor Operation, Rev. 16

LS-AA-104-1000, Exelon 50.59 Resource Manual, Rev. 3
CC-AA-112, Temporary Configuration Changes, Rev. 10

Drawings
205217

Notifications
20290150, 20294215

Orders
80090205, 80075675

Other Documents
S-C-SA-MSE-0925, 10 CFR 50.59 Evaluation of SC.OP-PM.SA-0001, Removal of Station Air Compressors from Service for Maintenance, Rev. 1
10 CFR 50.59 Review and Safety Evaluation of SC.OP-SO.SA-0002, Operating Temporary Air Compressors - Station Air, Rev. 0
Regulatory Change Process Determination of SC.OP-SO.SA-0002, Operating Temporary Air Compressors - Station Air, Rev. 10
S-C-CA-MDC-0526, Station Air Systems Air Receivers Design Basis & Capacity Verification, Rev. 1
SC.DE-BD.CA-0001, Salem Generating Stations - Control Air System, Rev. 0
DE-CB.CA-0014, Configuration Baseline Documentation for Control Air and Station Air Systems, Rev. 5
S-C-CA-MDC-1639, Air Load Management Program (ALMP) Update, Rev. 2
S-C-CA-MDC-0462, Control Air System Load Study, Rev. 1
PIRS 00960115125, Improperly Made Modification to Plant Design, circa June 1999
NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Rev. 3

Section 1EP6: Drill Evaluation

Notifications
20292151

Orders
70059867

Other Documents
Simulator Examination Scenario Guide LOR-022A
Salem Event Classification Guide

Section 4OA2: Identification and Resolution of Problems

Procedures
1-EOP-LOPA-1, Loss of All AC Power, Rev. 23
1-EOP-TRIP-1, Reactor Trip or Safety Injection, Rev. 24

1-EOP-APPX-6, RCP Seal Cooling Restoration, Rev. 22
2-EOP-APPX-6, RCP Seal Cooling Restoration, Rev. 22
2-EOP-LOCA-3, Transfer to Cold Leg Recirculation, Rev. 26
2-EOP-LOPA-1, Loss of All AC Power, Rev. 24
2-EOP-LOPA-2, Loss of All AC Power Recovery / SI Not Required, Rev. 22
2-EOP-LOPA-2, Post LOCA Cooldown and Depressurization, Rev. 22
2-EOP-LOPA-3, Loss of All AC Power Recovery / SI Required, Rev. 23
DE-PS.ZZ-0001(Q)-A3-SSA, Salem Fire Protection Report - Safe Shutdown Analysis, Rev. 18
DE-PS.ZZ-0001(Q)-A3-SSAR(003), Safe Shutdown Manual Action Feasibility Assessment, Rev. 1
DE-PS.ZZ-0001(Q)-A3-SSAR(007), Salem Fire Protection Report - Safe Shutdown Analysis, 460/230V Switchgear Room, Rev. 1
DE-PS.ZZ-0001(Q)-A3-SSAR(059), Salem Fire Protection Report - Safe Shutdown Analysis, Control Room Complex & Unit 1 Relay Room, Rev. 0
ER-AA-310-1001, Maintenance Rule - Scoping, Rev. 2
S1.IC-CC.RC-0016(Q), RCP Seal Temperature, Rev. 0
S1.OP-AB.RCP-0001(Q), Reactor Coolant Pump Abnormality, Rev. 13
S1.OP-AB.FIRE-0002(Q), Fire Damage Mitigation, Rev. 2
S1.OP-SO.CVC-0023(Q), CVCS Cross-Connect Alignment to Unit 2, Rev. 5
S2.OP-AB.CC-0001(Q), Component Cooling Abnormality, Rev. 11
S2.OP-AB.CR-0002(Q), Control Room Evacuation Due to Fire in the Control Room, Relay Room, 460/230V Switchgear Room, or 4kV Switchgear Room, Rev. 22
S2.OP-AB.CVC-0001(Q), Loss of Charging, Rev. 3
S2.OP-AB.FIRE-0001(Q), Control Room Fire Response, Rev. 2
S2.OP-AB.LOOP-0001(Q), Loss of Off-Site Power, Rev. 17
S2.OP-AB.RC-0004(Q), Natural Circulation, Rev. 6
S2.OP-AB.RCP-0001(Q), Reactor Coolant Pump Abnormality, Rev. 19
S2.OP-AB.SW-0005(Q), Loss of All Service Water, Rev. 3
SC.ER-DG.ZZ-0002, System Function Level Maintenance Rule Scoping Vs. Risk Reference for CVC, Rev. 2

Drawings
205328, Salem Unit 2 Chemical & Volume Control Operation

Notifications
20046891, 20140647, 20171205, 20187099, 20213439, 20239137, 20242343, 20250431, 20270913, 20280564, 20292672, 20295815, 20295829, 20295842, 20295843, 20295877, 20295884, 20295969, 20296007, 20296120, 20296127, 20296128, 20296129, 20296130, 20296264, 20296265, 20296266, 20296295, 20296296, 20296297,

Orders
30075839, 30086773, 60013439, 60037502, 60064981, 60065122, 70043154, 70048461, 70050029, 70053987, 70056354

Other Documents
10CFR50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
CV-1, CVCS Training P&ID, Rev. 2
Information Notice 05-14, Fire Protection Findings on Loss of Seal Cooling to Westinghouse Reactor Coolant Pumps
Operations Department Night Orders, Friday September 1, 2006
PN-1, Hot Shutdown Panel, Training P&ID, Rev. 0
RCP-1, RCP Seal Package, Training P&ID, Rev. 2
S1CVC-1TE155A, 11 CVC Reactor Coolant Pump Seal Water Inlet Temperature Data Sheet
S-C-CBV-MEE-1979, Containment Pressure/Temperature Response with RCP Seal Leakage During Control Room Fire, Rev. 0
S-C-FP-FEE-1746, Acceptable Operator Response Times to Appendix R Failures, Rev. 0
S-C-ZZ-NEE-0839, Time Analysis for Alternative Shutdown Capability For an Appendix R Fire Scenario, Rev. 1
WCAP-10541, Westinghouse Owners Group Report Reactor Coolant Pump Seal Performance Following a Loss of All AC Power
WCAP-10541, Supplement 1, High Temperature Extrusion Qualification Testing of Seals Eastern 7228A O-Ring Compound, Rev. 2
WCAP-16396-NP, Westinghouse Owners Group Reactor Coolant Pump Seal Performance for `Appendix R Assessments, Rev. 0
Westinghouse Technical Bulletin (TB)-04-22, Reactor Coolant Pump Seal Performance and Appendix R Compliance, Rev. 0
Westinghouse Technical Bulletin (TB)-04-22, Reactor Coolant Pump Seal Performance and Appendix R Compliance, Rev. 1
PSEG Metrics for Improving the Work Environment, Salem and Hope Creek Generating Stations, Quarterly Report, July 27, 2006

Section 4OA3: Event Followup

Procedures
S1.OP-SO.115-0012, 1B Vital Instrument Bus UPS System Operation, Rev. 12
SH.OP-AP.ZZ-0108(Q), Operability Assessment And Equipment Control Program, Rev. 19
S1.OP-AB.RCP-0001, Reactor Coolant Pump Abnormality, Rev. 19
SC.OP-DG.ZZ-0101, Salem Post-Trip Data Collection Guidelines, Rev. 8
S2.OP-SO.CVC-0001, Charging, Letdown, and Seal Injection, Rev. 27
S2.OP-SO.CVC-0012, CVCS Demineralizer - Normal Operations, Rev. 26
S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load, Rev. 26

Drawings
211370, 309945, 203007, 211365

Notifications
20291480, 20291577, 20291578, 20231322, 20233095, 20236992, 20233096, 20298370, 20298382, 20298384, 20298395, 20298412, 20298441

Orders
60064079, 50000059, 70016196

Other Documents
Salem Unit 1 Narrative Log;
Salem Unit 2 Technical Specification 3.4.11 Structural Integrity;
Salem Unit 2 Narrative Log;
Salem Unit 2 Manual Reactor Trip Prompt Investigation Report (NOTF 20298370);
Adverse Condition Monitoring Plan for RCP seal leakoff flows during low RCS boron concentration;
Data trends for pressurizer relief tank level, temperature, and pressure.
Data trends for reactor coolant pump seal return flow rates, bearing temperatures, and seal water temperatures;
Data trends for reactor coolant drain tank level;
Data trends for volume control tank temperature, level, and pressure;
Electric Power Research Institute (EPRI) description of increasing seal leakoff during operation due to primary chemistry changes.

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
EDG	Emergency Diesel Generator
MR	Maintenance Rule
NCV	Non-cited Violation
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PI	Performance Indicator
PSEG	Public Service Enterprise Group Nuclear LLC
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SAC	Station Air Compressor
SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
SWP	Service Water Pump
SWS	Service Water Strainer
UFSAR	Updated Final Safety Analysis
VIB	Vital Instrument Bus
WO	Work Order