March 26, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer and President
PSEG Nuclear LLC - X04
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: NRC INSPECTION REPORT 05000272/2000-011, 05000311/2000-011

Dear Mr. Keiser:

On February 10, 2001, the NRC completed an inspection of your Salem 1 and 2 reactor facilities. The enclosed report documents the results of that inspection. The preliminary findings were discussed in an exit meeting on February 27, 2001, with PSEG Nuclear management led by Mr. K. Davison and Mr. L. Wagner of your staff.

This 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 inspectors identified three issues of very low safety significance (Green) that were determined to involve violations of NRC requirements. The findings related to corrective actions on a service water valve, corrective actions for ventilation damper deficiencies, and the modification of a component cooling water pump impeller. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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, and the NRC Resident Inspector at the Salem facility.

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Sincerely,

***/RA/***

Glenn W. Meyer, Chief,
Projects Branch 3
Division of Reactor Projects

Enclosure: Inspection Report 05000272/2000-011, 05000311/2000-011

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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| Docket Nos: | 50-272, 50-311 |
| License Nos: | DPR-70, DPR-75 |
| Report No: | 05000272/2000-011, 05000311/2000-011 |
| Licensee: | PSEG Nuclear LLC |
| Facility: | Salem Nuclear Generating Station, Units 1 & 2 |
| Location: | P.O. Box 236<br>Hancocks Bridge, NJ 08038 |
| Dates: | December 31, 2000 - February 10, 2001 |
| Inspectors: | Fred L. Bower, (Acting) Senior Resident Inspector<br>F. Jeff Laughlin, Resident Inspector<br>K. M. Jenison, Senior Projects Engineer, DRP<br>Gregory C. Smith, Senior Physical Security Inspector, DRS<br>R. L. Fuhrmeister, Senior. Reactor Engineer, DRS<br>Leonard S. Cheung, Senior Reactor Engineer, DRS<br>K. A. Young, Reactor Engineer, DRS |
| Approved By: | Glenn W. Meyer, Chief,<br>Projects Branch 3<br>Division of Reactor Projects |

## Summary of Findings

IR 05000272-00-11, IR 05000311-00-11, on 12/31/2000 - 2/10/2001, Public Service Electric Gas Nuclear LLC, Units 1 and 2. Maintenance Rule Implementation, Operability Evaluations, Post Maintenance Testing and Other Areas (Cross-Cutting Issues).

The inspection was conducted by resident inspectors, a regional projects inspector and regional fire protection, radiation, and security specialists. This inspection identified three green findings which were also non-cited violations. The significance of most of the findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

**Cornerstone: Mitigating Systems**

- GREEN: PSEG Nuclear did not take timely and effective corrective actions for vibration induced failures of service water valve 21SW127 on the 21 component cooling heat exchanger outlet. Failure of 21SW127 rendered the 21 component cooling water (CCW) loop inoperable when the redundant 22 CCW loop was inoperable to support a liquid radwaste discharge.

  The finding had a credible impact on safety, since the failure of 21SW127 resulted in both component cooling water loops being out-of-service and resulted in a loss of the decay heat removal safety function. The SDP evaluation determined that the event was of very low safety significance due to the short duration of the event. The failure to implement effective, timely corrective actions for 21SW127 was considered a non-cited violation. (Section R12)

- GREEN: During maintenance on the 12 component cooling water (CCW) pump, impeller filing removed mass from the impeller, changed internal impeller and volute dimensions and clearances, changed the impeller blade angle, and changed the pump capacity. The filing was conducted with a maintenance order rather than the modification process, as required by station administrative procedures.

  The issue had a credible impact on safety, since this modification of a safety-related component outside of the approved modification process presented the risk that undocumented, unanalyzed, or procedurally uncontrolled dimensional changes could have degraded pump performance characteristics, such as net positive suction head, starting current or pump runout with their attendant impact on safety. Since only the mitigating system cornerstone was affected and this is a design qualification deficiency, the finding is considered to be of very low safety significance. Failure to control the filing of the 12 CCW pump impeller in accordance with administrative procedure requirements was considered a non-cited violation. (Section R19)

**Cornerstones: Public Radiation Safety**

- GREEN. Effective and timely corrective actions for ventilation damper deficiencies were not taken and resulted in low auxiliary building exhaust air flow due to a fire damper failing closed. Several previous similar events had demonstrated the susceptibility of damper locking devices.

  The finding had a credible impact on safety because radiological effluents could have bypassed the auxiliary building ventilation charcoal filter during accident conditions and resulted in a public radiation hazard. The finding was of very low safety significance because the auxiliary building was maintained at negative pressure during the low flow condition and any radioactive release would have been through high efficiency filtration via the normal monitored flowpath. The failure to take timely and effective corrective actions to preclude the recurrence of damper deficiencies was considered a non-cited violation. (Section 1R15)

B. <u>Licensee Identified Violations</u>

Violations of very low significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions, taken or planned by the licensee, appeared reasonable. These violations are listed in section 40A7 of this report.

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# Report Details

## SUMMARY OF PLANT STATUS

Unit 1 began the period at 100 percent power and remained there until the end of the period except for minor power reduction to support planned maintenance.

Unit 2 began the period at 100 percent power and remained there until January 12 when operators reduced power to 42 percent for main turbine valve testing and various scheduled plant maintenance activities. Operators restored the unit to full power on January 14 where the unit generally remained until the end of the period.

## 1. REACTOR SAFETY
**(Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity and Public Radiation Safety)**

R05 Fire Protection

a. Inspection Scope

The inspectors toured the following risk-significant areas to assess the material condition of fire protection systems and features and their operational line-up. Conditions related to the control of combustible material and ignition sources, and the operational status of fire barriers were observed. The inspectors also verified that any fire impairments were documented for equipment that was out of service and that appropriate compensatory measures were in place. Minor deficiencies were discussed with PSEG Nuclear personnel who took the appropriate corrective actions.

- Unit 1 122 foot elevation
- Unit 1 100 foot elevation around the containment entry hatch
- Unit 2 4160 Volt Switchgear Room
- Unit 2 460 Volt Switchgear Room
- Unit 1 Relay Room
- Unit 1 Service Water Bays
- Unit 2 Service Water Bays
- Unit 1 Emergency Diesel Generator Rooms
- Unit 2 Emergency Diesel Generator Rooms

b. Findings

No findings of significance were identified.

R07 Heat Sink Performance

a. Inspection Scope

The inspectors verified that processes and programs were adequate to ensure proper performance of the component cooling water and residual heat removal heat exchangers. A selected sample of corrective action documents, surveillance documents and the below listed engineering calculations were reviewed. Selected samples of inspection and cleaning methods and frequencies, and associated performance test results were also reviewed. The inspectors verified that: (1) the inspection, cleaning and testing results were recorded, evaluated and dispositioned; (2) the final heat exchanger conditions were acceptable; and, (3) the results were consistent with the assumptions made in the Updated Final Safety Analysis Report (UFSAR).

S-C-SW-MDC-1829, *Service Water System Heat Exchange Models*
S-C-CBV-MDC-1637, *Containment Fan Cooler Design Basis Capacity*
S-1-CC-MDC-1817, *Component Cooling Water System Hydraulic Analysis*

b. Findings

No findings of significance were identified.

R12 Maintenance Rule Implementation

.1 21 and 22 Component Cooling Heat Exchangers Out-of-Service

a. Inspection Scope

On January 24 control room operators identified that the 21 component cooling heat exchanger (CCHX) temperature had risen approximately 10 degrees above its normal operating point. Investigations by operations personnel identified that the 21 CCHX outlet service water (SW) air-operated valve, 21SW127, had failed closed. Control room operators declared the 21 component cooling water (CCW) loop inoperable. The 22 CCW loop had previously been declared inoperable to support a liquid radwaste release. Therefore, operators entered Technical Specification (TS) 3.0.3 because both redundant CCW loops were inoperable. The operators were able to restore the 22 CCW loop to an operable status and exit TS 3.0.3 within approximately one half hour. The inspectors reviewed the maintenance effectiveness associated with the failure of 21SW127. The review included interviews with engineering personnel and a walkdown of the 11, 12, 21 and 22 CCHXs. The inspectors also reviewed problem identification and resolution activities associated with the 21SW127 failure.

b. Findings

PSEG Nuclear's field walkdown of 21SW127 found that three supply and control air pressure gauges located in the valve positioner were damaged and leaking. PSEG Nuclear noted that there had been three other incidents of temporary erratic 21 CCHX control in January 2001 (January 10, 11, and 23) and also identified four Notifications (CM990328125, 20001287, 20035086 and 20035175) for failures of 21SW127 dating back to July 1998. A corrective action evaluation (Order 70008683) performed under Notification 20035175 concluded that the apparent cause of the 21SW127 failures was cavitation that had introduced a fatigue mechanism into the valve and actuator assembly causing premature failure of the gauges in the positioner. PSEG Nuclear concluded that there was no indication that the January 24 event was any different from previous events.

The inspectors also reviewed Notification 20035175 and Order 70008683 which were initiated July 17, 2000. The inspectors noted that the July 17, 2000, failure had not been evaluated to determine whether it was a functional failure. The initial PSEG Nuclear evaluation determined that an investigation should be completed to determine whether the anti-cavitation trim that was removed from the valve should be reinstalled, whether a design change was needed, or whether a replacement valve should be installed. The due date for this investigation was December 31, 2001. Additional PSEG Nuclear evaluations for these corrective action documents, performed subsequent to January 24, 2000, identified three additional Notifications (20036503, 20036472, and 20036386) for similar failures.

The inspectors reviewed the corrective action system and found the following twelve additional Notifications and Orders for 21SW127: 20036622, 60014249, 20036064, 70009001, 20035878, 60010708, 20035877, 60010541, 20017010, 70003716, 20010864 and 60003704. Notification 20036064 was initiated July 27, 2000, to evaluate vibration and gauge failures that occurred on 21SW127 on July 17 and 22, 2000. A proposed corrective action was to complete a 50.59 evaluation that would allow replacing the gauges with plugs. At the time of the January 24, 2001, failure, neither the 50.59 evaluation nor the plug installation had been completed. A second corrective action specified immediate adjustment of the 21SW127 hanger and vibration monitoring in accordance with Order 60014249. This order had not been completed and was scheduled for May 21, 2001.

This issue has a credible impact on safety, since the failure of 21SW127 resulted in both component cooling water loops being out-of-service. The CCHXs are required to support the residual heat removal heat exchangers; therefore, the inoperability of the CCHXs credibly affected the mitigating system cornerstone and resulted in a loss of the decay heat removal safety function. The Phase 1 significance determination process (SDP) analysis indicated that a Phase 2 SDP analysis was required. Since the Phase 2 SDP workbooks for Salem are not final, a Phase 3 SDP analysis was needed.

A regional senior reactor analyst (SRA) performed the Phase 3 analysis with the NRC PRA model for Salem, SPAR Rev. 3. The SRA ran the model for both CCHXs out-of-service for 1 hour. The dominant sequence was a small break loss of coolant accident (SBLOCA) with a failure to isolate the leak. Due to the short duration of the event, the change in core damage frequency (delta CDF) was ~ 4E-8. Therefore, the finding is considered to be of very low safety significance (Green). The inspectors determined that the corrective actions identified to be taken for numerous previous 21SW127 failures were untimely and ineffective to prevent recurrence of the failure of 21SW127. Failure to implement effective, timely corrective actions to resolve the condition constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions. However, because of the very low safety significance of the finding and because PSEG Nuclear has included this item in their corrective action program (Notification 20054964), this corrective action violation is being treated as a Non-cited Violation **(NCV 311/2000011-01)**, in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

.2 22 CCHX Temperature Control Service Water Valve, 22SW127

a. Inspection Scope

On January 4 control room operators identified that the SW air-operated valve, 22SW127, on the 22 CCHX outlet failed closed and declared the 22 CCW loop inoperable. Repairs were made to the 22 CCHX air-operated temperature and flow control valves under Notification 20052088. On January 3 this notification had been initiated to address 22 CCHX temperature swings. The inspectors reviewed the following Notifications associated with this issue: 20052088, 20052686, 20052692, and 20052724.

The inspectors reviewed the Unit 2 SW system health report (period 7/1/2000 to 9/15/2000), the list of functional failures and (a)(1) goals, and system level indicators applicable to the Unit 2 SW system to assess compliance with the maintenance rule. The inspectors observed that the system has been classified as red and in maintenance rule (a)(1) status due to exceeding the unavailability and functional failure performance criteria. The inspectors verified that additional functional failure goals have been established specific to air-operated valves.

Additional documents reviewed included: SE.MR.SA.02, *Maintenance Rule System Function Level and Risk Reference*; SH.SE-DG.ZZ-007(Z), *Preventable and Repeat Preventable System Functional Failure Determination*; NC.NA-AP.ZZ-0016(Q), *Monitoring the Effectiveness of Maintenance*; and SAEP 2001-01, *Salem Expert Panel Meeting Minutes*.

b. Findings

No findings of significance were identified.

.3 Units 1 and 2 Component Cooling Water System

a. Inspection Scope

Based on performance-based problems involving the 21 and 22 CCHXs (Report Sections 1R12.1 and 1R12.2), the inspectors reviewed the Unit 1 and Unit 2 CCW systems to assess the effectiveness and implementation of the maintenance rule program. The reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) safety significance classifications; (3) 10 CFR 50.6 (a)(1) and (a)(2) classifications; (4) the appropriateness of performance criteria; and, (5) recent performance trends. The inspectors reviewed the Units 1 and 2 CCW system health reports for the third (July 1 - September 15) and fourth (September 15 - December 31) quarters of the year 2000. The inspectors verified that the CCW system is a high risk significant and safety-related system that is in the scope of the maintenance rule. The inspectors also verified that both the Unit 1 and Unit 2 CCW systems are in (a)(2) status.

b. Findings

No findings of significance were identified.

.4 Unit 1 Service Water System

a. Inspection Scope

Based on the performance-based problems involving the 21 and 22 CCHXs (Report Sections 1R12.1 and 1R12.2), the inspectors reviewed the Unit 1 SW system to assess the effectiveness and implementation of the maintenance rule program. The reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) safety significance classifications; (3) 10 CFR 50.65 (a)(1) and (a)(2) classifications; (4) the appropriateness of performance criteria; and, (5) recent performance trends. The inspectors also reviewed the Unit 1 SW system health report report (period 7/1/2000 to 9/15/2000), the list of functional failures and (a)(1) goals, and system level indicators applicable to the Unit 1 SW system. The inspectors observed that the system has been classified as red and in maintenance rule (a)(1) status due to exceeding the unavailability and functional failure performance criteria. The inspectors verified that the SW system is a high risk significant and safety-related system that is in the scope of the maintenance rule. The inspectors verified that additional functional failure goals have been established specific to air-operated valves.

b. Findings

No findings of significance were identified.

R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

On January 4 the outlet SW control valve for the 22 CCHX failed closed and rendered the 22 CCW loop inoperable. The inspectors reviewed the Unit 2 control room narrative and technical specifications logs, the Unit 2 risk assessment evaluations for work weeks 156 and 001, and verified that the immediate risk assessments (performed by operations personnel) and follow-up risk assessments (performed by probabilistic risk assessment personnel) were performed in accordance with procedure SH.OP-AP.ZZ-0027(Q), *On-Line Risk Assessment*. Repairs to restore the 22 CCW loop were conducted under Order 60014665. The completed work order package was reviewed with PSEG Nuclear personnel. The inspectors verified that minor issues associated with the adequacy and completeness of the work package documentation and records were also entered into the corrective action program (Notifications 20053293, 20054124, 20052692 and 20055928).

b. Findings

No findings of significance were identified.

R15 Operability Evaluations

.1 Unit 1 Auxiliary Building Ventilation Low Flow

a. Inspection Scope

On January 6, 2001, PSEG Nuclear personnel determined that both auxiliary building ventilation (ABV) exhaust trains were inoperable due to low air flow following a design change to the 12 ABV exhaust fan. The inadvertent closure of fire damper 1ABF13 resulted in less than the technical specification (TS) required ABV exhaust air flow. Additionally, the closure caused reduced air flow from the safety injection pump room, residual heat removal pump rooms, and pipe chase area in the auxiliary building, which could have enabled an unfiltered release of radioactivity during accident conditions.

The inspectors reviewed operator actions, including entry into a 24-hour TS action statement, 3.7.7.1.a, for the low exhaust air flow and submission of a 10CFR50.72 report to the NRC for potentially inadequate filtration of effluent radiological material. The inspectors reviewed documentation, interviewed operations and engineering personnel, and walked down the ABV system to verify system operability.

b. Findings

On January 6 PSEG Nuclear conducted Unit 1 ABV air flow testing following the completion of a design change to the 12 ABV exhaust fan. The test determined that the exhaust air flow was low (64% of design, with 100%+/- 10% required). Further trouble-shooting revealed that the 1ABF13 fire damper was closed, which increased the air flow through a parallel ventilation branch excess flow damper 1ABS8, causing it to close as well. Operators subsequently opened the 1ABF13 damper, which caused the 1ABS8 damper to open, and restored satisfactory ABV flow (105% of design).

PSEG Nuclear determined that the apparent cause of the 1ABF13 closure was a locking wing nut on the lever arm, which holds the damper in the open position, loosened and allowed the damper shaft to rotate and drift closed. The wing nut apparently vibrated loose during normal system operation. Fire protection personnel had completed a functional test of this fire damper on November 17, 2000, using procedure S1.FP-ST.FBR-0028(Q), *Class 1 Fire Damper Operability Test.* The damper was left in the open position after this test. Therefore, the damper had closed sometime between the November functional test and the January 6, 2000, ABV air flow test.

The inspectors determined that corrective actions for two previous, recent occurrences concerning loose damper wing nuts could have precluded this event. The first occurrence was on July 14, 1999, when normally open fire damper, 1ABF11, was found in the closed position (Notification 20000694) which resulted in low ABV exhaust air flow. The second occurrence was on June 6, 2000, and involved the potential for ABV balance dampers to change position due to the locking wing nuts vibrating loose as documented in Notification 20033461. In addition, the inspectors noted that Licensee Event Report (LER) 05000354/2000-009 on Hope Creek documented a May 25, 2000 event in which the loosening of the mechanical locking device on a ventilation damper caused a system failure.

Corrective actions for the July 1999 event were narrowly focused. Corrective actions for the June 2000 event did not consider extent of condition (i.e., fire dampers) and had not been completed as of the January 2001 event. The inspectors concluded that these events provided earlier indication of loose locking wing nuts on dampers for which corrective actions were not completed in a timely manner.

The inspectors noted that multiple factors had contributed to the inadvertent closure of fire damper, 1ABF13, and had delayed detection of the damper closure. First, the indication for damper closure, which alarms on an annunciator panel in the control room back panel area, had been out of service since March 1999. This was the only remote indication of fire damper closure. The damper indication alarm repair had been scheduled for July 2000, but was postponed due to the need for a design change to access the alarm limit switch in the ventilation duct work. Secondly, operators were not aware that the damper alarm was out of service, did not hang a deficiency tag on the inoperable alarm panel window, and had not established any compensatory measures to ensure that the damper remained open.

The 1ABF13 fire damper closure caused the excess flow damper, 1ABS8, to close and resulted in inadequate ABV exhaust air flow. This event had a credible impact on safety, in that a containment barrier was degraded and radiological effluents could have bypassed the ABV charcoal filter, potentially resulting in radiation doses to the public in excess of regulatory limits during design basis accident conditions. Therefore, this finding was evaluated using the Public Radiation Safety and Reactor Safety SDPs. Since the auxiliary building and the affected rooms were under negative pressure at all times, and all the ABV system releases were through high efficiency particulate air filters and received normal effluent monitoring, the finding was considered to be of very low safety significance (Green). This finding was also determined to be Green using the Reactor Safety SDP, since it involved the radiological barrier function for the auxiliary building.

The inspectors concluded that the 1ABF13 closure could reasonably have been prevented by timely and effective corrective actions for the two previous, similar events. The failure to take effective, timely corrective actions was a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, which requires that adverse conditions be identified and corrected to preclude recurrence. However, because of the very low safety significance and because PSEG Nuclear documented it in their corrective action program (Notification 20052731), this corrective action violation is being treated as a Non-Cited Violation (**NCV 272/2000-011-02**), in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

.2 Operability Determinations and Follow-up Assessments

a. Inspection Scope

The inspectors reviewed selected operability determinations and follow-up assessments affecting risk significant mitigating systems to assess: (1) technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were appropriately addressed with respect to their collective impact on continued safe plant operation; and (4) where compensatory measures were involved, whether the measures were in place, would work as intended, and were appropriately controlled. Procedure SH.OP-AP.ZZ-0108(Q), *Operability Assessment and Equipment Control Program*, was used as a reference during the review of the Condition Resolution Operability Determinations (CRODs) and Follow-up Assessments (CRFAs). The CROD and CRFA evaluations associated with the following orders were reviewed:

- Order 70013148 — Containment Fan Coil Units (CFCUs) High Flow
- Order 70013041 — Emergency Diesel Generator and Cable Loading for 21-23 Auxiliary Building Ventilation Exhaust Fans
- Order 70013081 — Auxiliary Building Normal Area Air Flow Low

Regarding the CFCUs and Order 70013148, the inspectors verified that additional log readings of river temperature are being taken to ensure that further corrective actions are implemented prior to the river water temperature exceeding 70 degrees.

b. Findings

No findings of significance were identified.

.3 Unit 1 - A Train Containment Fan Coil Units

a. Inspection Scope

Due to recent failures of the containment fan coil units (CFCUs), a sample of Unit 1 A train CFCU's post maintenance testing records was reviewed to determined if adequate corrective actions and operability evaluations were being conducted. A sample of the testing accomplished under procedure S1.CBV-1VHE230, *Containment Fan Cooler Flow Test*, was reviewed. The test records and associated operability determinations were reviewed to determine if adequate scope, acceptance criteria and results were in compliance with the applicable technical specification (TS), equipment design bases and surveillance program criteria.

b. Findings

No findings of significance were identified.

.4 22 CCHX Outlet Valve (22SW127) Calibrated Incorrectly

a. Inspection Scope

The inspectors reviewed Use-As-Is Disposition, DRSE 80022389, 22SW127 Valve Calibrated Incorrectly with PSEG Nuclear personnel. This evaluation concluded that the 22SW127 valve was operable and acceptable to use-as-is until the next component cooling loop outage. The valve positioner had been calibrated using outdated setpoint information while the controlled equipment database was out of service. The inspectors verified that this issue had been correctly entered into the corrective action program (Notification 20052686) and that a tracking item had been initiated to ensure that 22SW127 is correctly calibrated during the next 22 component cooling water loop outage (Notification 20052724). Additional discrepancies were documented in Notification 20052951.

b. Findings

No findings of significance were identified.

R19 Post Maintenance Testing

.1 12 Component Cooling Water Pump

a. Inspection Scope

The inspectors reviewed the post maintenance testing (PMT) associated with design change package (DCP)1EC-344, 12 *Component Cooling Water Pump Flow Orifice*. Test documents were reviewed to determine the scope, acceptance criteria and results were in compliance with the applicable TSs, the UFSAR, design bases, and surveillance program criteria. The test methodology, documentation, technical justifications and engineering support were also reviewed for acceptability. Operability and functionality determinations were evaluated against the most recent system test data and design basis documentation.

b. Findings

Materials related to the PMT of the 12 component cooling water (CCW) pump indicated that impeller filing removed mass from the impeller, changed internal impeller and volute dimensions and clearances, and changed the impeller blade angle. The filing was performed under Order 70010521, rather than the station modification process. The filing modifications were performed without detailed instructions, did not control and include specific as-left dimensional acceptance criteria, and were not documented in enough detail to ensure that vendor specified internal dimensions were met. PSEG Nuclear procedure, NC.NA-AP.ZZ-0008(Q), *Control of Design and Configuration Change, Tests and Experiments*, requires equipment modifications that meet the definition of a design change to be processed as such. The procedure defines a design change as a change that affects the design requirements of form, fit and function including capacities and capabilities, physical sizes and dimensions for systems, structures and components. PSEG Nuclear failed to establish: (1) that adequate controls to ensure that explicit vendor specifications were incorporated into the pump impeller modifications; (2) that the supporting analysis addressed the impact of the modifications on the dimension and design parameters of the pump; or, (3) that the impact of the modification did not affect the ability of the pump to meet its post accident design function.

This issue has a credible impact on safety, since this modification of a safety-related component outside of the approved modification process presents the risk that undocumented, unanalyzed, or procedurally uncontrolled dimensional changes degraded pump performance characteristics such as net positive suction head, starting current or pump runout with their attendant impact on safety. Since this is a design qualification deficiency and steady state pump testing indicated that the pump had an increased flowrate with decreased vibration, the finding is considered to be of very low safety significance (Green). Failure to control the filing of the 12 CCW pump impeller in accordance with Salem administrative procedure requirements for design control is a violation of 10 CFR 50, Appendix B, Criterion III, *Design Control*. However, because of the very low safety significance of the finding and because PSEG Nuclear has included this item in their corrective action program (Notification 20053550), this corrective action violation is being treated as a Non-cited Violation **(NCV 272/2000011-03)**, in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

.2 Post Design Change and Maintenance Order Testing

a. Inspection Scope

The inspectors reviewed the post maintenance testing associated with the below listed design changes (DCP) and maintenance orders. Testing was reviewed to determine the scope, acceptance criteria and results were in compliance with the applicable technical specifications, the UFSAR, design bases, and surveillance program criteria. The test methodology, documentation, technical justifications and engineering support were also reviewed for acceptability. Operability and functionality determinations were evaluated against the appropriate most recent system test data and design basis documentation.

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| Orders 80006696, 80006726, 80006697 and 80006698 | 11, 12, 21 & 22 AFW Pump Runout Protection |
| Order 80005323 | 12 Radiation Monitor Steam Generator Blowdown |
| Order 80004253 | 12B Diesel Generator Starting Air |
| DCP 1EE411 and Order 80018352 | 11 and 23 AFW Pumps Pressure Indication |
| DCP 1EA1262 | AFW Pump Turbine Governors |
| DCP 1EC3519 and 2EC3437 | Auxiliary Feedwater Flow Requirements |

Additional documents reviewed included: instrument setpoint Calculation SC-AF002-01, *AFW Storage Tanks Level*; design Calculation S-C-AF-MDC-0445, *Auxiliary Feedwater System Hydraulic Analysis*; TS 3.7.1.3; and UFSAR Section 10.4. During the review of these documents, the inspector noted minor, licensee-identified discrepancies between these licensing and design bases documents that had not been entered into the corrective action system. The inspector verified that PSEG Nuclear took action to include this item in their corrective action program (Notification 20054454).

b. Findings

No findings of significance were identified.

R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the results from portions of the below listed surveillance tests. Test results, acceptance criteria, methodology, documentation, and engineering support were verified to be in compliance with the applicable TS acceptance criteria, equipment design bases, the UFSAR and surveillance program criteria. Operability and functionality determinations were evaluated against the most recent system test data and design basis documentation.

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| POs 4500086542 & 4500086540 | Main Steam Safety Valves Lift Test for N91590-00-0002, N55100-02-0029 & N55100-02-0039 |
| S2.OP-ST.CVC-003(Q) | 21 Charging Pump Inservice Testing |
| S1.IC-ST.SSP-0013(Q) | Reactor Trip Breaker “B” |
| S1.OP-ST.SF-000 & S2.OP-ST.SF-0002 | Spent Fuel Pool Cooling Pump Flow Test |

b. Findings

No findings of significance were identified.

**3. SAFEGUARDS**
**(Cornerstone: Physical Protection [PP])**

PP4 Security Plan Changes

a. Inspection Scope

An in-office review was conducted of changes to the Physical Security Plan, identified as Revision 14, and submitted to the NRC on July 14, 2000. The review of the plan revision confirmed that the changes were made in accordance with 10 CFR 50.54(p), and did not decrease the effectiveness of the Plan.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES [OA]

OA1 Performance Indicator Verification

.1 Units 1 and 2 -- Radiological Effluent Occurrences

a. Inspection Scope

The inspectors reviewed the following documents to verify that PSEG Nuclear met all reporting requirements for the Radiological Effluent Technical Specification and Offsite Dose Calculation Manual (RETS/ODCM) Radiological Effluent Occurrences performance indicator, from the third quarter 1999 to the fourth quarter 2000 (6 quarters):

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and
- associated procedures.

b. Findings

No findings of significance were identified.

.2 Units 1 and 2 -- Emergency Preparedness Performance Indicators

a. Inspection Scope

The inspection of the following performance indicators was conducted in conjunction with the inspection activities associated with the NRC-evaluated exercise at Hope Creek Generating Station and was documented in NRC Inspection Report 05000354/2000-004.

- Drill and Exercise Performance (DEP)
- ERO Drill Participation
- Alert and Notification System Reliability

b. Findings

No findings of significance were identified.

.3 Units 1 and 2 -- Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors verified the accuracy of and methods used to determine the performance indicator (PI) on *Reactor Coolant System (RCS) Specific Activity*. The inspectors reviewed the database of dose equivalent Iodine analysis results. The database was reviewed and compared against the Units 1 and 2 PI data submitted for the months of April through December 2000. The inspectors also observed the sampling and analysis of reactor coolant for dose equivalent Iodine, in accordance with procedures SC.CH-SA.RC-0222(Q), *Sampling Reactor Coolant and RHR Heat Exchanger Outlet* and NC.CH-RC.ZZ-2525(Q), *Gamma Spectroscopy Analysis Using CAS*. In addition, the inspectors verified that PSEG Nuclear documented minor deficiencies in the corrective action program (Notifications 200555595 and 20050512).

b. Findings

No findings of significance were identified.

.4 Units 1 and 2 -- Reactor Coolant System Leakage

a. Inspection Scope

The inspectors verified the accuracy of and methods used to calculate the PI on *Reactor Coolant System Leakage.* The inspectors verified the PI data submitted through review of the applicable page in the Units 1 and 2 TS surveillance data sheets for S1.OP-ST.RC-0008(Q) and S2.OP-ST.RC-0008(Q), *Reactor Coolant System Water Inventory Balance*. The sample of data reviewed included the months of April, July, October and December 2000.

b. Findings

No findings of significance were identified.

OA3 Event Follow-up

**(Closed) LER 05000272/01-001-00**: *Auxiliary Building Ventilation System Fire Damper Found Out of Position Limiting the Ability of the System to Perform Its Safety Function*. This event is documented in Section R15 of this report. The LER provided information consistent with the inspector's review of the event and was closed.

OA6 Management Meetings

.1 Exit Meeting Summary

On February 27, 2001, the inspectors presented their overall findings to members of PSEG Nuclear management led by Mr. K. Davison and Mr. L. Wagner of Salem Operations. PSEG Nuclear management acknowledged the findings presented and did not contest any of the inspectors' conclusions, although there were subsequent discussions regarding how the Reactor Oversight Program addressed the two cross-cutting issue findings. Additionally, they stated that none of the information reviewed by the inspectors was considered proprietary.

.2 PSEG Nuclear/NRC Management Meeting

On January 26, members of PSEG Nuclear senior management were in the Region I offices to meet with NRC management. Mr. H. Keiser, D. Garchow, M. Bezilla, and T. O'Conner represented PSEG Nuclear. Mr. H. Miller, R. Blough, and G. Meyer represented the NRC. The topic discussed included PSEG Nuclear reorganization plans and plant performance improvement initiatives.

OA7 Licensee Identified Violations

The following finding of very low significance was identified by PSEG Nuclear and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-cited Violations (NCV):

**NCV 272/2000-011-04**: 10 CFR 50, Appendix B, Criterion III, *Design Control*, requires that measures be established for the selection and review for suitability of application of parts that are essential to the safety-related functions of structures, systems and components. PSEG Nuclear identified that while repairing Unit 1 auxiliary building ventilation excess flow damper, S1ABV-1ABS8 under Order 60014824, a spring that assists in damper actuation was modified without the proper engineering documentation. This was documented in the corrective action program as Notification 20052729.

# SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Mark Bezilla, Vice President - Technical Support
David Garchow, Vice President - Operations
Gabor Salamon, Manager - Licensing/PSA
John Robertson, Operations Manager
Steven Mannon, System Engineering Manager
Frank Soens, Assistant Operations Manager
Larry Wagner, Work Management Department Lead
Terry Cellmer, Radiation Protection Manager
Robert Olsen, Assistant Operations Manager

b. List of Items Opened, Closed, and Discussed

**Opened/Closed**

| | | |
|---|---|---|
| 311/2000-11-01 | NCV | Untimely and ineffective corrective actions to prevent the recurrence of vibration induced failures of the 21 CCHX outlet valve, 21SW127. (Section 1R12) |
| 272/2000-11-02 | NCV | Untimely and ineffective corrective actions for ventilation damper deficiencies which resulted in low auxiliary building exhaust air flow due to a closed fire damper. (Section 1R15) |
| 272/2000-11-03 | NCV | The 12 component cooling water pump impeller was under-filed during maintenance without appropriate design controls. (Section 1R19) |
| 272; 311/2000-11-04 | NCV | Licensee identified that safety-related ventilation damper spring was modified during maintenance without appropriate design controls. (Section OA7) |

**Closed**

| | | |
|---|---|---|
| 272/01-001-00 | LER | Auxiliary Building Ventilation System Fire Damper Found Out of Position Limiting the Ability of the System to Perform Its Safety Function. (Section OA3) |

c. List of Acronyms

| | |
|---|---|
| ABV | Auxiliary Building Ventilation |
| AFW | Auxiliary Feedwater |
| AFWST | Auxiliary Feedwater Storage Tank |
| CCW | Component Cooling Water |
| CFCU | Containment Fan Coil Unit |
| CCHX | Component Cooling Heat Exchanger |
| CRFA | Condition Resolution Follow-up Assessments |
| CROD | Condition Resolution Operability Determinations |
| DCP | Design Change Package |
| LER | Licensee Event Report |
| NCV | Non-cited Violation |
| NRC | Nuclear Regulatory Commission |
| PARS | Publicly Available Records |
| PI | Performance Indicator |
| PMT | Post Maintenance Testing |
| PSEG | Public Service Electric Gas |
| RCS | Reactor Coolant System |
| SBLOCA | Small Break Loss of Coolant Accident |
| SDP | Significance Determination Process |
| SRA | Senior Reactor Analyst |
| SW | Service Water |
| TARP | Transient Analysis Response |
| TS | Technical Specification |
| UFSAR | Updated Final Safety Analysis Report |

# ATTACHMENT 1
# NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

| Reactor Safety | Radiation Safety | Safeguards |
|---|---|---|
| • Initiating Events | • Occupational | • Physical Protection |
| • Mitigating Systems | • Public | |
| • Barrier Integrity | | |
| • Emergency Preparedness | | |

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.