September 20, 1999

Mr. Harold W. Keiser
President and Chief Nuclear Officer
PSEG Nuclear LLC
Post Office Box 236
Hancocks Bridge, NJ 08038

Subject: NRC INSPECTION REPORT 50-272/99-06, 50-311/99-06

Dear Mr. Keiser:

On August 6, 1999, the NRC completed a team inspection of the design and performance capability of the auxiliary feedwater (AFW) system at your Salem 1 & 2 reactor facilities. The enclosed report presents the results of that inspection. The preliminary findings were discussed with Messrs. D. Garchow and F. Sullivan on August 6, 1999, and in several subsequent telephone conversations concluding on September 1, 1999.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission=s rules and regulations and with conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The team concluded that the AFW system was capable of performing its safety function under design basis conditions. Based on the results of this inspection, the NRC has determined that three violations of NRC requirements have occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Appendix F of the Enforcement Policy due to their low safety significance. The NCVs involved: (1) the failure to incorporate adequate acceptance criteria into AFW pump surveillance tests to ensure minimum design requirements would be maintained; (2) three examples of inadequate corrective action where non-conforming conditions were not promptly incorporated into your corrective action program; and (3) a design control failure regarding the qualification of the AFW flow control valve positioners. These NCVs are described in the subject inspection report.

The team also reviewed the safety system unavailability for the high pressure safety injection, auxiliary feedwater, emergency AC power and residual heat removal system performance indicators (PIs) that you submitted for the first six months of 1999. The safety system unavailability PI data was reported accurately.

In accordance with 10 CFR 2.790 of the NRC=s ARules of Practice,@ a copy of this letter and its enclosure will be placed in the NRC Public Document Room. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Salem Generating Station.

Sincerely,

ORIGINAL SIGNED BY:

Wayne D. Lanning, Director
Division of Reactor Safety

Enclosure: Inspection Report 50-272/99-06 and 50-311/99-06

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U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No:	50-272 & 50-311
License No.	DPR-70, DPR-75
Report No.	50-272/99-06, 50-311/99-06
Licensee:	PSEG Nuclear LLC
Facility:	Salem Nuclear Generating Station, Units 1 & 2
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	July 19 - August 6, 1999 August 9 - August 19, and September 1,1999 (Region I - Inoffice)
Inspectors:	L. Prividy, Senior Reactor Engineer, DRS P. Kaufman, Senior Reactor Engineer, DRS F. Arner, Reactor Engineer, DRS L. Scholl, Senior Reactor Engineer, DRS S. Pindale, Reactor Engineer, DRS H. Nieh, Resident Inspector, Salem 1 & 2, DRP
Approved By:	Lawrence T. Doerflein, Chief Engineering Programs Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Salem Generating Station, Units 1 & 2
NRC Inspection Report 50-272/99-06, 50-311/99-06

The report includes the results of a team inspection of the auxiliary feedwater system by region based inspectors.

Inspection findings were assessed according to potential risk significance, and were assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while not necessarily desirable, represent little risk to safety. WHITE findings would indicate issues with some increased risk to safety, and which may require additional NRC inspections. YELLOW findings would be indicative of more serious issues with higher potential risk to safe performance and would require the NRC to take additional actions. RED findings represent an unacceptable loss of margin to safety and would result in the NRC taking significant actions that could include ordering the plant to shut down. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

Cornerstone: Mitigating Systems

- Green. The acceptance criteria in the auxiliary feedwater (AFW) pump surveillance test procedures was inadequate since it did not ensure that the pump performance was capable of meeting accident analysis performance or pump operability criteria found in the Unit 1 and 2 Technical Specifications. This issue was considered potentially significant due to the allowable degraded core cooling capability associated with the established AFW pump minimum performance acceptance criteria. The issue was considered green in the significance determination process since it did not have an immediate impact on AFW system operability as determined by an evaluation of recent pump performance. Also, the issue has been entered into PSEG Nuclear=s corrective action program (CAP). The failure to establish AFW pump surveillance acceptance criteria, which ensured that each pump achieved its minimum design required performance, was a violation of NRC test control requirements. **(NCV 50-272/99-06-01; 50-311/99-06-01)** The team noted that the non-conforming condition, whereby surveillance test acceptance criteria was inconsistent with the AFW flow model pump performance assumptions, was not promptly incorporated into PSEG Nuclear=s CAP. This failure was a violation of NRC requirements concerning corrective action. **(NCV 50-272/99-06-02; 50-311/99-06-02)** (Section 1R21.1)

- Green. The team found that the positioners for the AFW flow control valves had been maintained with parts that were not qualified for use in safety-related applications. This issue was considered significant since the functionality for valves in a risk significant system was being challenged. The qualification issue was considered green in the significance determination process because it did not have an immediate impact on AFW system operability based on an engineering evaluation and the issue was being addressed by PSEG Nuclear=s CAP. This issue was considered to be a violation of NRC design control requirements. **(NCV 50-272/99-06-03; 50-311/99-06-03)** Although PSEG Nuclear entered the issue into the CAP, the team found that this action was not timely and was a violation of NRC requirements concerning corrective action. **(NCV 50-272/99-06-02; 50-311/99-06-02)** (Section 1R21.1)

- Green. The team identified some instances where PSEG Nuclear did not effectively implement the CAP.

  1. A prior performance weakness resulted in a governor control oil system needle valve setting for the 23 AFW pump turbine that was not optimal for actual operating conditions.

  2. During operation of the 23 AFW pump, the team identified that the turbine outboard bearing temperature exhibited an increasing trend, which represented an operability question, and PSEG Nuclear was not aware of this potential problem. Consequently, this problem was not entered into the CAP.

  3. Other examples of ineffective implementation of the CAP were identified, including deficiencies not promptly entered into the CAP or not adequately corrected.

  These issues were considered to be significant in that they indicated an overall CAP weakness since problems were noted in not identifying corrective action items and the lack of timeliness and effectiveness of corrective actions. The issues were considered green in the significance determination process because they did not have an immediate impact on AFW system operability and were ultimately included in PSEG Nuclear=s CAP. The issues collectively represented the third example of a violation of NRC requirements concerning inadequate corrective action. **(NCV 50-272/99-06-02; 50-311/99-06-02)** (Section 1R21.2)

Performance Indicator Verification

- The team reviewed the safety system unavailability for the high pressure safety injection, auxiliary feedwater, emergency AC power and residual heat removal system performance indicators (PIs) for the first six months of 1999 and found that the data was accurately reported for these systems and met the PI reporting guidance. (Section 4OA2)

1. REACTOR SAFETY
Cornerstone: Mitigating Systems

1R21 Safety System Design and Performance Capability

Introduction

The auxiliary feedwater (AFW) system at Salem Units 1 and 2 was reviewed using Inspection Procedure 71111, Attachment 21. AFW was selected since it is a risk significant mitigating system for responding to transients, such as station blackout, loss of offsite power, and loss of main feedwater.

.1 Design - Mechanical, Electrical, and Instrumentation and Controls

a. Inspection Scope

The team reviewed the AFW design and licensing basis documents to determine the system functional requirements during abnormal and accident conditions. For the documents reviewed, which included calculations and analyses, the team verified that the assumptions were appropriate, that proper engineering methods and models were used and that there was an adequate technical basis to support the conclusions. Where possible, the team performed independent calculations to evaluate the document adequacy. The review was performed to determine that: (1) the design basis was in accordance with the licensing commitments and regulatory requirements; (2) the design output documents such as drawings and procurement specifications were correct; and, (3) the installed system and components were tested to verify the design bases were met.

The team reviewed the Updated Final Safety Analysis Report (UFSAR) to establish the design and licensing basis for the AFW and interfacing systems. The piping and instrumentation drawings, the configuration baseline documents and the installed configuration were also reviewed to assess the capability of the system to satisfy the design intent. The adequacy of surveillance testing to ensure that adequate flow would be supplied to the steam generators during worst case accident conditions was also reviewed.

The team reviewed changes made to the system to verify that the system met the design and licensing basis in the modified configuration and that the changes did not introduce any unreviewed safety questions.

b. Observations and Findings

The team identified two issues that could have resulted in the inoperability of important AFW equipment. The acceptance criteria in the AFW pump surveillance test procedures was inadequate since it did not ensure that the pump performance was capable of meeting accident analysis performance or pump operability criteria found in the Unit 1 and 2 Technical Specification (TS). Also, the team found that the positioners for the AFW flow control valves had been maintained with parts that were not qualified for use in safety-related applications, thus rendering them as potentially inoperable. These potential inoperability issues were considered significant due to (1) the degraded core cooling capability associated with less than required pump performance; and (2) challenging the functionality of the control valves in a risk significant system. The issues were considered green in the significance determination process because they did not have an immediate impact on AFW system operability as determined by evaluations of recent pump performance and control valve operation. Also, the issues were being addressed by PSEG Nuclear=s corrective action program.

<u>Pump Design Requirements</u>

Technical Specification 4.7.1.2.b.1 requires that each motor driven auxiliary feedwater pump (MDAFWP) be capable of developing a discharge pressure of 1275 psig on recirculation. Technical Specification 4.7.1.2.b.2 requires that the turbine driven auxiliary feeedwater pump (TDAFWP) develop a discharge pressure greater than or equal to 1500 psig on recirculation flow when secondary steam generator pressure is greater than 680 psig. PSEG Nuclear had verified these requirements during testing in accordance with Section XI, AInservice Testing,@ of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The team noted that in Information Notice 97-90, AUse of Nonconservative Acceptance Criteria in Safety-related Pump Surveillance Tests,@ the NRC alerted licensees that if minimum acceptable design performance was more stringent than the ASME Code acceptance criteria, then the test acceptance criteria must be adjusted to avoid the actual pump performance from being allowed to degrade below the minimum acceptable design performance.

The AFW system design for the motor driven pumps incorporates a run-out protection feature to ensure cavitation is minimized under certain accident scenarios, such as faulted steam generators and reduced steam generator back pressure conditions. The system resistance is varied by throttling a flow control valve as a function of the pump discharge pressure. For the 11, 12 and 21 MDAFWPs, a range of 1150 to 1350 psig will throttle the AFW flow control valve from full closed to full open. Since the capability of the 22 MDAFWP is about 150 psi lower at the design point flow of 440 gpm than the other pumps, its control range is from 1000 psig to 1200 psig. The team noted that the allowable 7.5% to 10% reduction in pump performance permitted by the surveillance test procedures would result in the runout protection logic inadvertently sensing the need for flow control valve throttling, when, in fact, the discharge pressure reduction would be due to pump degradation. The effect would be throttling of the flow control valve to the steam generator, resulting in a higher system frictional loss and less flow delivered by the AFW system. In summary, the existing system runout protection results in an additional reduction to system flowrate capability under degraded pump performance conditions. The team concluded that any allowable pump degradation should be modeled to ensure that minimum design flowrates can be maintained.

Pump Performance Testing Review

The design output of the AFW flow model contained in calculation D01.6-868 consisted, in part, of the minimum flow requirements for various accident scenarios such as loss of main feedwater and steam line break outside containment transients. The flow model simulated the pumps, piping, valves and other components in the system. The design pump curve (head versus flow) provided by the manufacturer was used to predict pump performance.

The team reviewed the AFW pump test results and found the performance of the 22 MDAFWP had degraded about 2.2% from the pump manufacturer=s performance curve. Independent calculations performed by the team indicated that the 22 MDAFWP might not develop the minimum flows included in the calculation of record. PSEG Nuclear reviewed the data and determined that the AFW system minimum flow performance documented in table 2.0 of calculation D01.6-868 would not be achieved by the 22 MDAFWP at the assumed steam generator backpressures of 1133 psia or the 1150 psig stated in the AFW bases section of the plant TS. PSEG Nuclear performed an operability determination which utilized best-estimate assumptions for steam generator back pressures along with performance data from the last five full flow pump tests and concluded that the 22 MDAFWP would achieve the minimum required flow of 440 gallons per minute (gpm). Similar test reviews indicated a 1.7% reduction in the 12 MDAFWP performance at full flow conditions. The team had no operability concern regarding the 12 MDAFWP since it had a higher developed head at the pump design point of 440 gpm.

PSEG Nuclear performed additional calculations to determine an acceptable level of pump degradation which would maintain minimum licensing bases flow assumptions. For the 22 MDAFWP, PSEG Nuclear concluded that a 3.8% allowable level of pump degradation would result in maintaining minimum accident analysis flow assumptions and would replace the existing allowable pump degradation (a nominal 9%) in the surveillance procedure. PSEG Nuclear indicated that similar calculations would be performed for the other MDAFWPs to determine acceptable levels of pump degradation.

Concerning the TDAFWPs, the team noted that current testing performed to satisfy TS requirements permits increasing the speed of the turbine until the 1500 psig discharge pressure TS criterion is satisfied. Thus, an indeterminate amount of degradation would be allowed since it could be compensated for by increasing pump speed until the required 1500 psig pressure was achieved at the recirculation flowrate of 400 gpm. The team reviewed the latest full flow test results for the 13 and 23 TDAFWPs and noted that the 23 pump was about 5.0% degraded from the performance curve developed in calculation S-C-AF-MDC-0430, Revision 0. The team determined that the most recent test data from a May 21, 1999 full flow test (S2.OP-AF-0007) supported the capability of the 23 pump to supply 880 gpm at the best estimate steam generator backpressures included in the 22 MDAFWP operability review noted above. However, the allowable pump degradations currently in the surveillance tests would not allow design TDAFWP flows to be achieved at the speed set for automatic startup (3450 rpm).

Pump Performance Testing Review - Conclusion

The team determined that the acceptance criteria in the AFW pump surveillance test procedures did not ensure that pump performance was capable of meeting accident analysis performance or pump operability criteria found in the AFW system bases section of the Salem Unit 1 and 2 Technical Specifications. Specifically, the procedure allowed a span of 7.5% to 10.0% pump performance degradation before considering the pumps to be inoperable or in the required action range. This allowable degradation was inconsistent with assumptions utilized within the PSEG Nuclear=s AFW flow model system analysis in that the flow model had not accounted for any pump performance degradation. Although PSEG Nuclear had established in-service testing (IST) acceptance criteria that met the requirements specified in the ASME Code, the criteria would have allowed the AFW pumps to degrade below the performance assumed in the accident analysis. This potential inoperability issue was considered significant due to the degraded core cooling capability associated with less than required pump perfomrance in the risk significant AFW system. The issue was considered green in the significance determination process since it did not have an immediate impact on AFW system operability as determined by an evaluation of recent pump performance. The failure to establish AFW pump surveillance acceptance criteria, which ensured that each pump achieved its minimum design required performance, was considered to be the first example of a violation of 10CFR Part 50, Appendix B, Criterion XI, ATest Control.@ This violation is being treated as a Non-Cited Violation (NCV), consistent with Appendix F of the NRC Enforcement Policy**. (NCV 50-272/99-06-01; 50-311/99-06-01)** PSEG Nuclear initiated notification number 20001328 in accordance with the Corrective Action Program (CAP) requirements to review the AFW pump technical specifications, develop a licensing change request to ensure AFW required pump performance is consistent with accident analysis assumptions, and modify AFW surveillance test acceptance criteria accordingly.

The team noted that the non-conforming condition, whereby surveillance test acceptance criteria was inconsistent with the AFW flow model pump performance assumptions, was not promptly incorporated into PSEG Nuclear=s CAP. Specifically, on July 23, 1999, the discrepancy had been identified to PSEG Nuclear personnel. However, notification 20001328 was not entered into the CAP until July 30, 1999, after further subsequent questions and independent NRC calculations identified concerns with the operability of the 22 MDAFWP on July 29, 1999. The team considered this lack of timeliness was significant, considering the challenge to operability of a risk significant system presented by the issue. This issue was considered green in the significance determination process since it had no immediate impact on AFW system operability. This issue was the first example of a violation of NRC requirements concerning corrective action. 10 CFR 50, Appendix B, Criterion XVI, ACorrective Action,@ requires that measures shall be established to assure conditions adverse to quality are promptly identified and corrected. In response to the discussion held on July 23, 1999, PSEG Nuclear viewed that the current level of pump performance supported design calculation minimum flowrates and actions had been initiated to re-analyze existing flow model calculations to determine acceptable levels of pump degradation. However, the team determined that the lack of a timely notification initiation for the identified discrepancy was inconsistent with the expectations documented within PSE&G Nuclear procedure, NC.WM-AP.ZZ-0000(Q), Revision 0, ANotification Process,@ which stated, in part, that Awhen in doubt, a notification should be processed.@ This violation is being treated as a Non-Cited

Violation, consistent with Appendix F of the NRC Enforcement Policy. **(NCV 50-272/99-06-02; 50-311/99-06-02)** PSEG Nuclear has included this issue appropriately into the CAP (Notification 20004451).

Air Operated Valve Positioners

The team noted that the AFW system design utilized air-operated components to control flow during automatic system operation and for remote manual operation of the flow control valves and the turbine driven pump governor and steam stop valves. The team also found that the design bases of the system considered the control air system to be highly reliable and assumed it to be available following an accident.

When responding to questions regarding the safety classification of control air system components, PSEG Nuclear informed the team that the positioners for the flow control valves had been maintained with parts that were not appropriate for use in safety-related applications. Based on an their initial review, PSEG Nuclear determined that the cause of the problem was the downgrading of the purchase class of certain piece parts during a procurement engineering change performed in 1995.

This issue was considered significant since the functionality of the valves in a risk significant system was being challenged. PSEG Nuclear performed an evaluation to assess the potential effect of the non-safety parts on the affected valves operability. This evaluation concluded that the affected valves were operable but that actions were necessary to restore the valves to full qualification as a safety-related component. The team evaluated this condition and concluded that, since the use of non-qualified parts did not affect the valve operability and the issue was being addressed by PSEG Nuclear=s CAP, this issue was green in accordance with the criteria of the significance determination process. PSEG Nuclear was also performing an extent of condition review to determine if there are any similar problems with other installations.

10 CFR 50, Appendix B, Criterion III, "Design Control", requires, in part, that measures be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions. It further requires that measures be established for the selection and review for suitability of application of materials, parts, equipment and processes that are essential to the safety-related functions of the structures, systems and components.

Contrary to the above, PSEG Nuclear had not ensured that the applicable design basis was correctly translated into specifications, drawings, procedures or instructions, and had not ensured that measures were established for the selection and review of materials, in that repair parts for positioners on the flow control valves were not properly selected and reviewed to ensure their suitability for performing a safety-related function. In accordance with Appendix F of the NRC Enforcement Policy, this violation is being treated as a Non-Cited Violation. PSEG Nuclear has entered this issue into the CAP (Notification #20002187). **(NCV 50-272/99-06-03;50-311/99-06-03)**

Although PSEG Nuclear entered the issue into the CAP, the team found that this action was not timely in that the issue was identified on July 23, 1999, but the corrective action notification was not initiated until August 3, 1999. The team considered that this lack of timeliness was significant, considering the challenge to operability of a risk significant system presented by the issue. The issue was considered green in the significance determination process since it did not have an immediate impact on AFW system operability. This was the second example of a violation of NRC requirements concerning corrective action. 10 CFR 50, Appendix B, Criterion XVI, ACorrective Action,@ requires that measures shall be established to assure conditions adverse to quality are promptly identified and corrected. This violation is being treated as a Non-Cited Violation, consistent with Appendix F of the NRC Enforcement Policy. **(NCV 50-272/99-06-02; 50-311-99-06-02)** PSEG Nuclear has included this issue appropriately into the CAP (Notification 20004451).

.2 Operations and Maintenance

a. Inspection Scope

The team reviewed a number of activities to verify that the AFW system was installed, operated and maintained consistent with the design and licensing bases. The operational standby readiness and material condition of the AFW system was assessed by conducting system walkdowns and reviewing procedures, operator logs, design and vendor documents, component maintenance history records, and system health reports. The team also interviewed licensed and non-licensed operators and engineers. As part of this review, the team evaluated a sample of licensee-identified problems in the CAP as well as some emergent problems to assess the effectiveness of PSE&G Nuclear=s corrective actions.

2. Observations and Findings

The team identified some instances where PSEG Nuclear did not effectively implement the CAP.

- A prior performance weakness resulted in a governor control oil system needle valve setting for the 23 AFW pump turbine that was not optimal for actual operating conditions.

- During operation of the 23 TDAFWP, the team identified that the turbine outboard bearing temperature exhibited an increasing trend, which represented an operability question, and PSEG Nuclear was not aware of this potential problem. Consequently, this problem was not promptly entered into the CAP.

- Other examples of ineffective implementation of the CAP were identified, including deficiencies not promptly entered into the CAP or not adequately corrected.

These issues were considered significant since they involved weaknesses in the CAP that negatively impacted the performance of risk significant equipment. Furthermore, the issues indicated an overall CAP weakness since problems were noted in not identifying corrective action items and the lack of timeliness and effectiveness of corrective actions. The issues were considered green in the significance determination process because they did not have an immediate impact on AFW system operability and were ultimately included in PSEG Nuclear=s CAP. The issues collectively represented the third example of a violation of NRC requirements concerning inadequate corrective action. 10 CFR 50, Appendix B, Criterion XVI, ACorrective Action@, requires that measures shall be established to assure conditions adverse to quality are promptly identified and corrected. This violation is being treated as a Non-Cited Violation, consistent with Appendix F of the NRC Enforcement Policy. **(NCV 50-272/99-06-02; 50-311/99-06-02)** PSEG Nuclear has included these issues appropriately into the CAP (Notification 20004451).

<u>23 TDAFWP Governor Needle Valve Setting</u>

PSEG Nuclear initiated performance of the quarterly surveillance test of the 23 TDAFWP on July 23. After about 45 minutes of operation, operators noticed a speed oscillation of about 200 rpm peak to peak. Also, one of the U-bolt type supports on the two-inch recirculation line broke during the test. Operators declared the pump inoperable pending further evaluation.

PSEG Nuclear attributed the cause for the speed oscillations to be related to the setting of the governor oil compensating needle valve, which affects the responsiveness of the governor. The needle valve was last adjusted during an overspeed test using compressed air rather than steam during the most recent refueling outage (May 1999). The turbine response is different when using compressed air as the turbine motive force. The use of compressed air resulted in a different position of the governor oil needle valve, causing the speed instability during the July 23 test. The needle valve is sensitive to oil viscosity changes as well as actual machine responsiveness. PSEG Nuclear subsequently evaluated the condition of the 23 TDAFWP and determined that the setting would not have resulted in a loss of governor function since the pump was able to deliver the required pressure and flow to perform its safety function. Although the team considered that this determination was reasonable, the issue was considered green in the significance determination process since the evaluation supporting the determination was largely dependent on engineering judgment (e.g., 200 rpm oscillation was acceptable while higher oscillations would not be). An inspection of the Unit 1 TDAFWP revealed its governor needle valve to be set correctly after its last overspeed test which was performed with compressed air. PSEG Nuclear had recognized that tuning the governor using compressed air in lieu of steam would result in a different needle valve adjustment; however, an activity was not scheduled and performed to implement the necessary adjustment. As additional corrective action, PSEG Nuclear issued Notifications 20002900 and 20002878 to add an activity to the governor actuator replacement recurring tasks (411001 and 461001) to require final tuning using steam as the motive force.

Turbine Outboard Temperature Trend

During operation of the 23 TDAFWP on August 4 to address oscillation problems previously experienced on July 23, the team requested a trend graph of its parameters to verify that the turbine governor had sufficient cooling. All parameters were normal with the exception of the turbine outboard bearing, which was about 165 °F and still increasing at the end of the one-hour run. The team questioned PSEG Nuclear regarding the ability of the 23 TDAFWP to meet its design function of operating for an extended time period considering there was a computer point high temperature alarm set at 180 °F for this bearing. PSEG Nuclear conducted an additional run of the pump on August 10 and documented a technical evaluation, which included vendor information indicating the bearing damage point to be 220 °F. This test demonstrated that a stable outboard bearing temperature was achieved at 164 °F. Although operability was not compromised, PSEG Nuclear did not initiate prompt actions to evaluate the August 4 temperature data that potentially degraded the design function of the 23 TDAFWP. The team considered that this lack of timeliness was significant, considering the challenge to operability of a risk significant system presented by the issue. The issue was considered green in the significance determination process because it did not have an immediate impact on system operability and was ultimately resolved by inplant testing.

Other Examples of Ineffective Implementation of the Corrective Action Program

The team identified two instances where PSEG Nuclear=s entries of deficiencies into the corrective action program were untimely. The 23 TDAFWP experienced excessive speed oscillations during a routine surveillance on July 23; however, this discrepancy was not entered into the CAP until July 30. The team considered that this lack of timeliness was significant, considering the challenge to operability of a risk significant system presented by the issue. The second instance involved the team=s identification on July 20 that a large (about 3 foot diameter) manway cover was inappropriately stored adjacent to a return line to the AFW storage tank, presenting a seismic interaction concern. Although PSEG Nuclear implemented prompt actions to remove the cover, the item was not entered into the CAP until August 5. The team considered that this lack of timeliness was significant, considering the challenge to operability of a risk significant system presented by the issue and specific NRC prompting to enter the problem into the CAP. Both instances were considered green in the significance determination process because they did not have an immediate impact on AFW system operability and the problems were ultimately included in PSEG Nuclear=s CAP.

The team also identified two instances of ineffective corrective actions concerning past deficiencies. In the first instance, the team reviewed action requests (ARs) 990105086, 990222151, and 990323164 related to freezing and water intrusion concerns for the Unit 2 AFW storage tank overflow line loop seal and identified that PSEG Nuclear failed to implement the recommended corrective actions identified in the three ARs. Secondly, AR 980502123 recommended that TDAFWP IST performance procedure changes be implemented to prevent operational problems with the AFW pump turbine trip valve (MS52). However, the team identified that additional procedures that provided detailed operation of the valve were not similarly revised. PSEG Nuclear subsequently initiated additional items in the CAP to address these problems. The team considered that these two instances were significant since they indicated an overall CAP weakness which affects actions on multiple risk significant systems. Both instances were considered green in the significance determination process because they did not have an immediate impact on AFW system operability.

.3 Surveillance and Testing

a. Inspection Scope

The inspectors reviewed test procedures and recent performance data to verify that the following AFW components met their design and licensing bases:

$ motor and turbine driven pumps
$ steam generator inlet control valves (AF21s) and recirculation valves (AF40s)
$ automatic actuation circuitry

b. Observations and Findings

The AF40 valves have a safety function to automatically close during emergency motor driven pump operation. This function limits pump recirculation flow to ensure that the required flow rates are delivered to the steam generators. Technical specification (TS) surveillance 4.7.1.2.c.1 requires verification that each automatic AFW valve not secured in position actuates to the correct position during an actual or simulated actuation signal. The corresponding test procedure, S1(2).OP-ST.AF-0009, APlant Systems - Auxiliary Feedwater,@ does not verify that the AF40 valves close at the specified pump discharge flow rate. However, PSEG Nuclear verified that the valves operate properly during quarterly inservice testing using a simulated signal, in addition to a periodic calibration of the associated flow instruments that send the actuating signal to the valves during operation. The team concluded that this approach adequately satisfied TS surveillance 4.7.1.2.c.1.

OTHER ACTIVITIES [OA]

4OA2 Performance Indicator Verification

1. Inspection Scope (IP 71151)

The team verified the accuracy and completeness of data used to calculate and report safety system unavailability for high pressure safety injection, auxiliary feedwater, emergency AC power and residual heat removal systems performance indicators (Pis) for both Salem units. To assess the reported PI data accuracy the team reviewed control room operating logs, condition reports, system health reports and interviewed the individuals compiling and trending the PI data.

2. Observations and Findings

From a review of safety system unavailability data for the first and second quarters in 1999 and through discussions with cognizant personnel, the team determined that the PI data was accurately reported for these safety systems using guidance contained in the Nuclear Energy Institute (NEI) draft PI guideline document, 99-02, revision B. None of the unavailability data reviewed for these systems exceeded the PI thresholds for increased regulatory response. With respect to the AFW system, the Unit 1 TDAFWP was unavailable for about 64 hours in January 1999 due to governor performance problems as well as human performance problems during the related testing activities. No PI threshold was exceeded as a result of this unavailability.

PI unavailability performance data deficiencies identified by PSEG Nuclear of previously submitted data for 1997 and 1998 was properly entered into the CAP. PSEG Nuclear plans to submit the revised PI unavailability data to the NRC.

4OA4 Other (IP 71152)

a. (Closed) IFI 50-311/97-16-03: Conformance of Calculations and Procedures with Technical Specification 4.7.6.1.d (5). The inspectors determined that previous discrepancies identified within calculation S-C-CAV-MDC-1569, AUnits 1 & 2 Control Room Envelope Cooling and Heating Load,@ and documented within inspection report 97-16, had no significant impact on the capability of the system to perform its design function. Surveillance procedures, S1(2).RA-ST.CAV-0001,@Control Room Emergency Ventilation System Surveillance Test,@ S1.RA-ST.CAV-0004(Q), AUnit 1 Control Room Emergency Air Conditioning System (EACS) Surveillance,@ and S2.RA-ST.CAV-0003(Q), AUnit 2 Control Room Emergency Air Conditioning System (EACS) Surveillance,@ were reviewed and found to adequately ensure the capability of the Control Room EACS to remove the assumed heat load in accordance with TS 4.7.6.1.d(5). No violations were identified with this issue.

b. (Closed) URI 98-09-04: weak resolution of carbon dioxide discharge valve seal failure. The inspectors reviewed and discussed the issues and corrective actions with PSEG Nuclear system engineering personnel. PSEG Nuclear=s evaluation of the valve seal failure determined it to be an isolated failure. Subsequent inspections of similar valves did not reveal any seal degradation. The inspectors concluded that PSEG Nuclear=s corrective actions were reasonable, and that no violation or deviation from regulatory requirements existed.

4OA5 Management Meetings

PSEG Nuclear representatives were informed of the purpose and scope of the inspection at an entrance meeting conducted on July 19, 1999. The team presented the preliminary inspection findings to Messrs. D. Garchow and F. Sullivan and other members of PSEG Nuclear management on August 6,1999, who acknowledged the findings presented. Several subsequent telephone conversations were held with PSEG Nuclear personnel to further discuss the inspection findings. The last conversation occurred on September 1, 1999. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Public Service Enterprise Group

T. Carrier	PSA Supervisor
D. Garchow	Vice President, Engineering
A. Garcia	AFW System Engineer
J. Grant	Salem Maintenance Manager
M. Kafantans	Salem Assistant Operations Manager
S. Mannon	Assistant Manager, Salem System Engineering
G. Nagy	Manager, Salem System Engineering
T. Ross	Nuclear Fuels Supervisor
G. Salamon	Manager, Licensing
F. Sullivan	Director, Nuclear Plant Engineering
E. Villar	Licensing Engineer
J. Zudans	Manager, Mechanical Design

New Jersey Department of Environmental Protection

R. Pinney	Nuclear Engineer

ITEMS OPENED AND CLOSED

Opened and Closed

50-272/99-06-01; 50-311/99-06-01	NCV	Inadequate acceptance criteria in surveillance test procedures
50-272/99-06-02; 50-311/99-06-02	NCV	Multiple examples of ineffective implementation of corrective action program
50-272/99-06-03; 50-311/99-06-03	NCV	In adequate design control concerning replacement parts for positioners for AFW flow control valves

Closed

50-311/97-16-03	IFI	Conformance of Calculations and Procedures with Technical Specification 4.7.6.1.d(5).
50-272/98-09-04; 50-311/99-09-04	URI	Weak Resolution of Carbon Dioxide Discharge Valve Seal Failure.

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
gpm	gallons per minute
IST	Inservice Testing
LLC	Limited Liability Corporation
MDAFWP	Motor Driven Auxiliary Feedwater Pump
NRC	Nuclear regulatory Commission
PI	Performance Indicator
PSEG	Public Service Enterprise Group
psia	pounds per square inch absolute
psig	pounds per square inch gauge
rpm	revolutions per minute
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VP	Vice President