November 8, 2001

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

SUBJECT: SALEM NUCLEAR GENERATING STATION - NRC INSPECTION REPORT 50-272/01-09, 50-311/01-09

Dear Mr. Keiser:

On September 30, 2001, the NRC completed an inspection of your Salem 1 & 2 reactor facilities. The enclosed report documents the inspection findings which were discussed on October 15, 2001, with Mr. T. O'Conner and other members of your staff.

The inspectors 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. Specifically, this inspection involved seven weeks of resident inspection and region-based inspections of the radiological protection and the licensed operator re-qualification programs. No findings of significance were identified.

Since September 11, 2001, PSEG Nuclear has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to PSEG Nuclear. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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.html (the Public Electronic Reading Room).

Sincerely,

/RA/

William A. Cook, Acting Chief
Projects Branch 3
Division of Reactor Projects

Enclosure: Inspection Report 50-272/01-09, 50-311/01-09
Attachment: Supplemental Information

Docket Nos. 50-272 & -311
License Nos. DPR-70 & -75

cc w/encl: E. Simpson, Senior Vice President and Chief Administrative Officer
M. Bezilla, Vice President -Technical Support
D. Garchow, Vice President - Operations
G. Salamon, Manager - Licensing
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N. Cohen, Coordinator - Unplug Salem Campaign
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Distribution w/encl: Region I Docket Room (with concurrences)
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	50-272, 50-311
License Nos:	DPR-70, DPR-75
Report No:	50-272/2001-09, 50-311/2001-09
Licensee:	PSEG Nuclear LLC
Facility:	Salem Nuclear Generating Station, Units 1 & 2
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	August 12 - September 30, 2001
Inspectors:	Raymond K. Lorson, Senior Resident Inspector Fred L. Bower, Resident Inspector Joseph T. Furia, Senior Health Physicist Paul H. Bissett, Senior Operations Engineer Todd H. Fish, Operations Engineer Travis Tate, Project Manager, NRR
Approved By:	Glenn W. Meyer, Chief, Projects Branch 3 Division of Reactor Projects

Summary of Findings

IR 05000272-01-09, IR 05000311-01-09, on 8/12 - 9/30/2001, Public Service Electric Gas Nuclear LLC, Salem Units 1 and 2. Resident inspector report.

The inspection was conducted by resident inspectors, a regional radiation specialist, and regional operations specialists. This inspection identified no findings of significance. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Findings

A violation of very low significance which was identified by PSEG Nuclear in Notification 2074507 and reviewed by the inspectors. Corrective actions, taken or planned by the licensee, appear reasonable. This violation is listed in section 40A7 of this report.

Report Details

SUMMARY OF PLANT STATUS

Unit 1 began the period at 100 percent power. On September 8, power was reduced to approximately 40 percent for scheduled main turbine system maintenance and testing. The unit was returned to full power on September 10, after completion of emergent repairs to a reheat steam valve. On September 24, operators reduced power to about 78 percent in response to an unexpected loss of the No. 2 station power transformer (Section R14). Subsequently, the operators manually tripped the unit in response to a decreasing main condenser vacuum and increasing main turbine back pressure condition. On September 27, the operators initiated a cooldown to Mode 5 to support replacement of a leaking pressurizer safety valve. The unit remained in Mode 5 for the duration of the period.

Unit 2 began the period at 100 percent power. On September 15, power was reduced to approximately 40 percent for scheduled main feed pump and main turbine control system maintenance, and main turbine valve testing. The unit was returned to full power on September 21 after completing planned and emergent maintenance activities on the feedwater system. On September 24, operators reduced power to about 73 percent in response to a loss of the No. 2 station power transformer (Section R14). The operators returned the unit to full power on September 25, where it remained for the duration of the period.

1. REACTOR SAFETY

Initiating Events, Mitigating Systems, and Barrier Integrity [Reactor - R]

R11 Licensed Operator Requalification

.1 Quarterly Resident Inspector Observation

a. Inspection Scope

On August 21, 2001, the inspectors observed two licensed operator annual requalification evaluated simulator scenarios to assess operator performance and evaluators' critiques. The first scenario involved the loss of the secondary heat sink and the restoration of feed flow. The inspectors observed the operating crew perform the following emergency operating procedures (EOP) and functional recovery procedures (FRP): EOP-TRIP-1, Reactor Trip or Safety Injection, EOP-FRHS-1, Response to Loss of Secondary Heat Sink, and EOP-Trip-2, Reactor Trip Response. The second scenario involved faulted steam generators and a stuck open safety valve. The inspectors observed the operating crew perform the following EOPs: EOP-TRIP-1, Reactor Trip or Safety Injection, EOP-LOSC-1, Loss of Secondary Coolant, and EOP-LOCA-1, Loss of Reactor Coolant. Following the simulator exercise, the inspectors observed the scenario evaluators, which included training instructors and operations management personnel, review, discuss, and critique the operators' performance during the scenarios.

b. Findings

No findings of significance were identified.

.2 Biennial Licensed Operator Requalification Program Review

a. Inspection Scope

A review was conducted of recent operating history documentation found in inspection reports, licensee event reports, PSEG Nuclear's corrective action program (CAP), and the most recent NRC plant issues matrix (PIM). The inspectors reviewed specific events from the CAP (i.e., Notifications), that indicated possible training deficiencies, to verify that they had been appropriately addressed.

The following inspection activities were performed using NUREG 1021, Rev. 8, "Operator Licensing Examination Standards for Power Reactors," Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program," Appendix A "Checklist for Evaluating Facility Testing Material" and Appendix B, "Suggested Interview Topics":

* The operating tests for the week of September 2, 2001 were reviewed for quality and performance.

* The results of the year 2000 biennial written exams and annual operating tests for years 2000 and 2001, to date, were reviewed for quality, performance, and grading. (An assessment of whether failure rates are consistent with the guidance of NUREG-1021, Revision 8, and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," was deferred until completion of the current requalification cycle).

* Observations were made of the dynamic simulator exams and job performance measures (JPM) administered during the week of September 2, 2001. These observations included facility evaluations of crew and individual performance during the dynamic simulator exams (two scenarios per crew), and individual performance of five JPMs.

* The remediation plan for a crew failure in the simulator was reviewed, as well as the remediation plans for an individual JPM failure and an individual written exam failure.

* One licensed operator license reactivation was reviewed.

* Operators were interviewed for feedback on their training program and the quality of training received.

* Simulator performance and fidelity were reviewed for conformance to the reference plant control room (Unit 2).

* A sample of records for requalification training attendance, program feedback, reporting, and medical examinations were reviewed for compliance with license conditions, including NRC regulations.

b. Findings

No findings of significance were identified.

R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed applicable documentation and system performance data for selected systems at Units 1 and 2 to confirm proper implementation of the maintenance rule. The following documents were reviewed: the Maintenance Rule System Functional Level and Risk Significance Guide (SE.MR.SA.01), procedure SH.ER-DG.ZZ-0001(Z), “Preventable and Repeat Preventable System Functional Failure Determination,” the preventable functional failure database, Notifications, and the recent system health reports. The following systems were reviewed at each unit:

- Auxiliary Feedwater.
- Residual Heat Removal.
- Containment Spray.

b. Findings

No findings of significance were identified.

R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For the selected maintenance activities listed below, the inspectors observed selected work activities and/or reviewed work plans to verify that: risk assessments were performed in accordance with SH.OP-AP.ZZ-0027(Q), “On-Line Risk Assessment”; risk assessment reviews for unscheduled availability were entered into the control room narrative logs; the risk associated with scheduled work activities was properly managed; and appropriate contingency plans were identified in the integrated work schedule.

- Emergent maintenance activities following the failure of the 15 Containment Fan Cooler Unit to start in low speed (Notification 20074431), and following identification that four containment spray system valves (11, 12, 21, and 22CS10) were not properly leak rate tested per 10 CFR 50 Appendix J requirements (notification 20074507).

- A planned maintenance outage conducted on the 11 safety injection pump during the week of September 17.
- The emergent activities performed to correct seat leakage past the 1PR5 relief valve.

b. Findings

A PSEG Nuclear-identified finding for the containment spray valve test issue was reviewed and is discussed in Section OA7. No additional findings of significance were identified.

R14 Personnel Performance During Nonroutine Plant Evolutions

The inspectors reviewed PSEG Nuclear's response to a September 24 event involving the unexpected loss of section 1 of the 500kv bus following the loss of the No. 2 station power transformer. This resulted in the loss of three circulating water (CW) pumps and entries into TS 3.8.1.1.a at both units. The operators initiated a load reduction at each unit to approximately 80 percent in accordance with the circulating water system operating abnormal procedure (AB.CW-0001) and the rapid load reduction abnormal procedure (AB.LOAD).

Unit 1 power was stabilized at 78 percent until the 13A CW pump tripped on high CW traveling screen differential pressure. Operators began a load reduction to 30 percent power and then manually tripped the Unit 1 reactor in accordance with AB.CW-0001 due to increasing condenser back pressure. In response to increasing condenser back pressure, operators reduced load and stabilized Unit 2 at 73 percent power.

The inspectors reviewed the applicable operating logs, selected plant data, TS requirements and the abnormal operating procedures to evaluate the operators' response to this event. The inspectors also reviewed the transient assessment response plan (TARP) team report (Notification 20078117) that was developed for this event to assess the adequacy of PSEG Nuclear's immediate and planned follow-up corrective actions. The inspectors also observed the presentation of the post-trip review report to the Station Operations Review Committee (SORC).

b. Findings

No findings of significance were identified.

R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected condition report operability determinations (CROD) 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 CRODs. The inspector reviewed CROD No. 70018521, "23 Service Water Ventilation Thermostat Controls."

b. Findings

No findings of significance were identified.

R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance test documentation and/or observed test activities for selected risk significant mitigating systems to assess whether the effect of the testing on the plant had been adequately addressed by control room and engineering personnel; the test scope was appropriate for the maintenance performed; acceptance criteria were clear and adequately demonstrated operational readiness, consistent with design and licensing basis documents; test instrumentation met calibration, range and accuracy requirements; tests were performed, as written, with applicable prerequisites satisfied; and that equipment was returned to the status required to perform its safety function. The following test activities were reviewed:

- Order 60021627, 15 CFCU Failed to Start in Low Speed, was retested using procedures S1.OP-ST.SW-0010(Q), Inservice testing Containment Fan Coil Unit (CFCU) Service Water Valves and S1.OP-ST.CBV-0003(Q), Containment Systems Cooling Systems.
- Order 60021749, 16 Service Water Pump Replacement and Vibration Troubleshooting, was retested using procedure S1.OP-ST.SW-0006(Q), Inservice Testing - 16 Service Water Pump.
- Order 50039274, 11 Safety Injection Pump was retested using procedure S1.OP-ST.SJ-0001(Q).

b. Findings

No findings of significance were identified.

R22 Surveillance Testing

.1 Surveillance Test Observations

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk-significant systems, structures, and components (SSCs) to assess whether the SSCs satisfied Technical Specifications, Updated Final Safety Analysis Report, and licensee procedure requirements. The inspectors assessed whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were witnessed:

- S1.OP-ST.AF-0002(Q), Inservice Testing - 12 Auxiliary Feedwater Pump.
- S1.OP-ST.SW-0006(Q), Inservice Testing - 16 Service Water Pump.

b. Findings

No findings of significance were identified.

.2 Review of Containment Air Temperature Surveillance Test Measurement

a. Inspection Scope

The inspectors reviewed section 3.4 of operations procedure S1.OP-DL.ZZ-0003 (Q), “Control Room Log - Modes 1-4,” that described the method for determining the average containment air temperature. This review was performed to determine whether the Technical Specification (TS) Surveillance Requirement 4.6.1.5 to verify that the containment average temperature was within its limit (i.e. less than or equal to 120°F) once every twenty four hours was properly implemented. The inspectors interviewed licensing and engineering personnel and reviewed applicable documentation including:

- TS 3/4.6.1.5 bases
- Updated Final Safety Analysis Report (UFSAR) Section 15.4
- Salem Unit 1 Containment Air Temperature Data from August 9, and August 10, 2001
- Engineering Calculation S-2-CBV-MDC-1738, revision 0, “Calculation of Containment Average Air Temperature - Salem Unit 2"
- Engineering Evaluation S-2-CBV-MEE-1115, revisions 0 and 1, “Salem Unit 2 - Containment Temperature Monitoring Design Basis”
- Engineering Evaluation S-C-CBV-MEE-1280, revision 0, “Salem Unit 1 and Salem Unit 2 Containment Temperature Monitoring Design Basis”
- Safety Evaluation for TS Amendments 195 and 178 that issued the current TS 4.6.1.5 requirement.

b. Findings

Operations procedure S1.OP-DL.ZZ-0003 (Q) directed the operators to determine the containment average temperature by computing the arithmetic average of the ten permanently installed temperature instruments located in the primary containment. The temperature instruments are located at fixed axial locations with three detectors located at each of three elevations (84', 106', and 136') and one detector located at the 121' elevation. The temperature in the containment is a function of the axial location and, as expected, the containment temperature is higher at the higher elevations. The volumes associated with each axial location vary in size, however, the axial zone from the 136' elevation to the top of the containment dome is the largest zone and comprises approximately 75% of the containment air volume.

Engineering evaluation EE-S-2-CBV-MEE-1115, revision 0, indicated that the TS limitations on the containment air temperature were to ensure that the overall containment average air temperature would not exceed the initial temperature condition assumed in the accident analysis for a steam line break or loss of coolant accident. This evaluation determined that the “best definition of the primary containment weighted average air temperature is a volume weighted sampling,” and the evaluation provided an approach to compute the average containment temperature using a volume weighted method. PSEG Nuclear subsequently developed engineering evaluations EE-S-2-CBV-MEE-1115, revision 1 and S-C-CBV-MEE-1280, revision 0 and determined that the volumetric weighted averaging approach specified in EE-S-2-CBV-MEE-1115, revision 0, yielded unrealistically high values and did not account for non-air structures located within the containment that could act as a “heat sink” during postulated accident conditions. The latter two engineering evaluations provided the basis for PSEG Nuclear to compute the containment average temperature using the arithmetic averaging approach.

The inspectors noted that the arithmetic averaging approach yielded less accurate and lower values than the volumetric weighted approach since the highest temperatures (i.e at the 136' elevation) were recorded at the zone with the largest air volume. For instance, using Unit 1 containment temperature data obtained from August 9 and August 10 the inspector estimated that the volume weighted average temperature would be several degrees higher than arithmetic average temperature.

PSEG Nuclear licensing personnel indicated that the arithmetic averaging method for determining the containment average air temperature met the TS 4.6.1.5 surveillance requirements since the surveillance requirement was intended to verify that the containment air and heat sink temperatures met the initial conditions assumed in the accident analyses rather than just the air temperature. The inspectors determined that further review was needed to determine whether PSEG Nuclear’s approach for determining the containment average air temperature met the applicable licensing requirements. This issue will remain unresolved pending further review. **(URI 50-272 and 311/2001-09-01)**

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation

a. Inspection Scope

On September 4, 2001, the inspectors observed an evaluated simulator scenario during a licensed operator requalification examination to assess senior licensed operator performance regarding emergency classification declarations. The inspectors observed that the scenario provided one opportunity for the Operations Superintendent (OS) to make an emergency classification declaration. The inspectors observed that the OS did not determine the correct classification. The inspectors verified that although this was not considered a critical task, the examination evaluators identified the incorrect classification. The inspectors verified that the unsuccessful drill and exercise performance (DEP) performance opportunity will be forwarded to the emergency planning and licensing staffs for inclusion in the third quarter calculation of the DEP indicator.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupation Radiation Safety [OS]

OS1 Access Control

a. Inspection Scope

The inspector reviewed exposure significant work areas, high radiation areas, and airborne radioactivity areas in the plant and evaluated associated controls and surveys of these areas to determine if the controls (i.e., surveys, postings, barricades) were acceptable. For these areas, the inspector reviewed radiological job requirements and attended job briefings to determine if radiological conditions in the work area were adequately communicated to workers through briefings and postings. The inspector also verified radiological controls, radiological job coverage, and contamination controls to ensure the accuracy of surveys and applicable posting and barricade requirements. The inspector determined if prescribed radiation work permits (RWPs), procedure and engineering controls were in place; whether licensee surveys and postings were complete and accurate; and if air samplers were properly located. The inspector conducted reviews of RWPs used to access these and other high radiation areas to identify the acceptability of work control instructions or control barriers specified. The inspector reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy and reviewed portions of the licensee’s training and qualifications program for radiation workers to ensure that their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities. The controls implemented by the licensee were compared to those required under plant technical specifications (TS 6.12) and 10 CFR 20, Subpart G for control of access to high and locked high radiation areas.

The inspector reviewed quality assurance monitoring feedback (QAMF) Nos. 2000-0163 and 2001-0234 to examine licensee actions for identification, documentation, and corrective actions related to the access control program. Lastly, the inspector also reviewed one licensee notification (No. 20076508) documenting an access control issue related to an unsecured radiologically controlled area (RCA) boundary.

b. Findings

No findings of significance were identified.

OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed work to be performed during the Spring 2001 refueling outage (1R14). Areas reviewed included the use of low dose waiting areas, a review of on-job supervision provided to workers and a review of individual exposures from selected work groups. An evaluation of engineering controls utilized to achieve dose reductions and analysis of licensee source term reduction plans was also conducted. The bases for the licensee's outage goal of not more than 125.3 person-rem, with the outage total being 133.6 person-rem (106.6% of the goal), was reviewed.

The inspector observed radiation worker and radiation protection (RP) technician performance during high dose rate or high exposure jobs to determine if workers demonstrated proper techniques to maintain occupational exposures as low as is reasonably achievable (ALARA) and if their training/skill level was sufficient with respect to the radiological hazards and the work involved. The primary job observed involved change out of a high dose rate filter from the spent fuel pool filtration and cooling system.

The inspector reviewed ALARA job evaluations, exposure estimates, and exposure mitigation requirements and compared ALARA plans with the results achieved. A review was conducted of the integration of ALARA requirements into work procedures and RWP documents, the accuracy of person-hour estimates and person-hour tracking, and generated shielding requests and their effectiveness in dose rate reduction.

A review of actual exposure results versus initial exposure estimates for current work was conducted including: comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, exposure report timeliness and exposure report distribution to support control of collective exposures to determine conformance with the requirements contained in 10 CFR 20.1101(b). The inspector reviewed the planning of five high exposure jobs performed during 1R14 and their associated ALARA packages including: reactor maintenance (estimate: 26.35 person-rem; actual: 28.3 person-rem); primary side steam generator work (estimate: 25.9

person-rem; actual: 26.7 person-rem); in-service inspection (estimate: 13.8 person-rem; actual: 13.9 person-rem); and, secondary side steam generator work (estimate: 6.4 person-rem; actual: 6.3 person-rem).

The inspector also reviewed QAMF No. 2001-0171 related to the access control program.

b. Findings

No findings of significance were identified.

OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed field instrumentation utilized by RP technicians and plant workers to measure radioactivity, including portable field survey instruments, friskers, portal monitors, and small article monitors. The inspector conducted a review of selected radiation protection instruments observed in the RCA, specifically verification of proper function and certification of appropriate source checks for those instruments that were utilized to ensure that occupational exposures are maintained in accordance with 10 CFR 20.1201. The inspector also reviewed records and made direct observation of portions of the licensee's respiratory protection program including calibration and maintenance of port-a-count testing apparatus and facilities utilized to clean, maintain, and test respirators.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

OA1 Performance Indicator (PI) Verifications

Cornerstone: Initiating Events

a. Inspection Scope

The inspectors verified the following performance indicators (PI) for the initiating events cornerstone for Units 1 and 2:

- Unplanned Scrams per 7000 Critical Hours.
- Scrams with Loss of Normal Heat Removal.
- Transients per 7000 Critical Hours.

The inspectors reviewed the monthly Operating Data Reports applicable to the four quarters of operation beginning with the third quarter of 2000 and ending with the second quarter of 2001 to determine the number of reactor critical hours. The inspectors reviewed licensee event reports, a sample of NRC inspection reports, and a sample of licensee quarterly power history reports to verify the number of reactor trips and unplanned transients that had occurred. The inspectors also independently calculated the reported values to verify their accuracy. The inspectors used procedure NC.LR-DG.ZZ-0001(Z), NRC Performance Indicator Submittal and Validation Process Desktop Guide.

b. Issues and Findings

No findings of significance were identified.

OA3 Event Follow-Up

.1 (Closed) LER 272/01-006-00: Reactor Trip due to a Degraded Termination on Phase "A" Neutral Current Transformer Field Wiring. This LER described a Unit 1 reactor trip event that was discussed in NRC Inspection Report 2001-07. No new information was identified and this LER is closed.

.2 (Closed) LER 311/01-003-00: Containment Spray Additive Tank Exceeded Technical Specification Limit Allowable Outage Time. This LER was a re-submittal of LER 311/01-002-00 (closed in NRC Inspection Report 2001-08) to differentiate it from Special Report 311/01-002. No new information was identified and this LER is closed.

.3 (Closed) Special Report 311/01-005: Inoperable R-15 Radiation Monitor. This report discussed a moisture intrusion problem that rendered the R-15 radiation monitor inoperable. PSEG Nuclear's corrective actions appeared reasonable and complete. This special report is closed.

.4 (Closed) LER 311/01-004-00: Missed Technical Specification Requirement for Containment Isolation Valve. This LER discussed a minor violation described in NRC Inspection Report 2001-08 involving the failure to de-energize the 2SJ4 and 2SJ5 valves during a maintenance activity that rendered the 2SJ12 valve containment isolation function inoperable. No new information was identified and this LER is closed.

OA6 Management Meetings

a. Exit Meeting Summary

On October 15, 2001, the inspectors presented their overall findings to members of PSEG Nuclear management led by Mr. T. O'Conner. PSEG Nuclear management did not indicate that any of the information reviewed by the inspectors was considered proprietary.

b. PSEG Nuclear/NRC Management Meeting

On September 7, senior PSEG Nuclear management including Mr. H. Keiser and Mr. T. O'Conner met with NRC Region I management including Mr. H. Miller and Mr. R.

Blough in the Region I office to discuss recent plant performance trends and improvement initiatives.

c. NRC Public Meeting with UNPLUG Salem

On August 16, the resident inspectors and other Region I staff participated in an NRC public meeting conducted by NRR to explain the reactor oversight process relative to the inspection of the Salem steam generators. The meeting was conducted in Pennsville, New Jersey and was open for public participation.

OA7 Licensee Identified Violations

The following finding of very low significance was identified by PSEG Nuclear and was a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dis-positioned as a non-cited violation (NCV). If you deny this non-cited violation, 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, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors at the Salem facility.

NCV Tracking Number	Requirement Licensee Failed To Meet
NCV 50-272/01-09-002	10 CFR 50, Appendix B, Criterion XVI, requires, in part, that conditions adverse to quality be promptly identified and corrected. Contrary to the above, on August 2, 2001, a PSEG Nuclear engineer identified that containment spray (CS) system valves (11, 12, 21, and 22 CS10) had not been properly leak rate tested as required by 10 CFR 50, Appendix J; however, an appropriate operability assessment and corrective actions were not implemented until August 13, 2001. This issue was entered into PSEG's corrective action process as notification 20074507.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

K. Davison, Operations Manager
D. Garchow, Vice-President, Operations
G. Salamon, Licensing Manager
L. Waldinger, Operations Director
T. Cellmer, Radiation Protection Manager
K. O'Hare, ALARA Superintendent
M. Hassler, Radiation Protection Operations Superintendent - Salem
T. Neufang, ALARA Supervisor

b. List of Items Opened, Closed, and Discussed

Opened

50-272&311/01-09-01	URI	Review of containment air temperature surveillance test measurement. (Section R22)

Opened/Closed

50-272/01-09-002	NCV	Containment spray system valves had not been properly leak rate tested as required by 10 CFR 50, Appendix J. (Section OA7)
50-272/01-006-00	LER	Reactor trip due to a degraded termination on phase "A" neutral current transformer field wiring. (Section OA3)
50-311/01-003-00	LER	Containment spray additive tank exceeded technical specification limit allowable outage time. (Section OA3)
50-311/01-004-00	LER	Missed technical specification requirement for containment isolation valve. (Section OA3)
50-311/01-005	Special Report	Inoperable R-15 radiation monitor. (Section OA3)

c. List of Acronyms

ALARA	As Low As Is Reasonably Achievable
CAP	Corrective Action Program
CFCU	Containment Fan Coil Unit
CFR	Code of Federal Regulations
CROD	Condition Report Operability Determination
CS	Containment Spray
CW	Circulating Water
DEP	Drill and Exercise Performance
EOP	Emergency Operating Procedures
FRP	Functional Recovery Procedures
JPM	Job Performance Measures
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OS	Operations Superintendent
PARS	Publicly Available Records
PI	Performance Indicator
PIM	Plant Issues Matrix
PSEG	Public Service Electric Gas
QAMF	Quality Assurance Monitoring Feedback
RCA	Radiologically Controlled Area
RP	Radiation Protection
RWP	Radiation Work Permit
SDP	Significance Determination Process
SORC	Station Operations Review Committee
SSCs	Systems, Structures and Components
TARP	Transient Assessment Response Plan
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report