U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report Nos. 50-272/88-13
50-311/88-13

License Nos. DPR-70
DPR-75

Licensee: Public Service Electric and Gas Company

P. O. Box 236

Hancocks Bridge, New Jersey 08038

Facility Name: Salem Nuclear Generating Station - Units 1 and 2

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: May 3, 1988 - June 6, 1988

Inspectors: R. W. Borchardt, Senior Resident Inspector
K. Halvey Gibson, Resident Inspector

Approved by: N F Dudley FOR 7-6-88
P. D. Swetland, Chief, Reactor Projects date
Section No. 2B, Projects Branch No. 2, DRP

Inspection Summary:
Inspections on May 3, 1988 - June 6, 1988 (Combined Report Numbers 50-272/88-13 and 50-311/88-13)

Areas Inspected: Routine inspections of plant operations including: followup on outstanding inspection items, operational safety verification, maintenance, surveillance, engineered safety feature walkdown, and review of licensee event reports.

Results: The failure to adequately complete a technical specification required surveillance test is cited in this inspection report (paragraph 7). Although licensee identified, this violation's similarity to a previous violation indicates that continued emphasis is necessary in the surveillance test area.

NRC review of the licensee identified reactor protection system calibration inadequacies was completed during this report period (paragraph 7). Based upon the low safety significance and the licensee's program for comprehensive dynamic testing of reator protection functions, this deficiency was classified as a licensee identified violation.

DETAILS

1. Persons Contacted

Within this report period, interviews and discussions were conducted with members of licensee management and staff as necessary to support inspection activity.

2. Followup on Outstanding Inspection Items (92701, 92703)

(Closed) Unresolved Item 50-272/82-34-01; Discrepancies in recording filler metal type and proper heat number. Inspection report 50-272/82-34 states that this item was resolved prior to issuance of the inspection report. No further inspection is required and this item is closed.

(Closed) Inspector Follow Item 50-272/84-09-01 and 311/84-09-01; QA audits of staff performance. This item is closed based on NRC QA inspection (87-31/87-32) and non-licensed staff training inspection (87-10/87-13) in which no related concerns were identified.

(Closed) Unresolved Item 50-311/84-45-02; Visual examination of painted surfaces. The inspector reviewed Deviation 1 to Procedure SWRI-NDT-900-1/50 which delineates when a VT examination over paint is allowable. This item is closed.

(Closed) Unresolved Item 50-311/84-45-03; NDE Qualification records. The inspector reviewed supplemental certification data and had no further questions at this time. This item is closed.

(Closed) IE Compliance Bulletin No. 86-03 (50-272/86-BU-03; 50-311/86-BU-03); Minimum flow recirculation line failures of ECCS pumps. The inspector reviewed licensee responses dated November 20, 1986 and January 15, 1987, in which the licensee concluded that the intent of GDC 35 is satisfied for the Safety Injection, Residual Heat Removal, and Charging/Safety Injection Systems. The inspector has no further questions at this time. This bulletin is closed for Salem Units 1 and 2.

(Closed) Inspector Follow Item 50-272 and 50-311/86-02-05; Gas decay tank sample calibration standard. The inspector verified that a gas calibration standard (in styrofoam media rather than gel) is used to calibrate the Johnson bomb geometry for gas decay tank sample analysis. The inspector also reviewed subsequent intercomparison results delineated in NRC combined Inspection Report Nos. 272/88-10 and 311/88-10. This item is closed.

(Closed) Inspector Follow Item 50-311/86-03-01; Comparison of licensee analytical results to BNL results of water samples. This item is closed based on analytical measurement data comparisons reported in NRC combined inspection reports 272/87-33; 311/87-34 and 272/88-11; 311/88-11.

(Closed) Inspector Follow Item 50-272/86-06-02; Engineering Evaluation for fire door No. 121-1. The evaluation of auxiliary building ventilation balance is an ongoing long term action. To ensure operability of fire door 121-1 until the ventilation problems are resolved, a new door has been installed and fire protection verifies the integrity of the door once per shift. This item is closed.

(Closed) Unresolved Item 50-272/86-11-03; Thimble tube wall thinning. Thimble tubes were replaced during the seventh refueling outage per design change package 1EC-2232. This item is closed.

(Closed) Inspector Follow Item 50-272/86-19-01; Heat shrinkable tubing test results. Raychem test results were provided to the licensee on October 16, 1986 and will be reviewed by NRC per TI 2500/17 (IEN 86-53). This item is closed.

(Closed) Inspector Follow Item 50-311/86-33-01; Computer RWP form cluttered. The inspector reviewed radiation protection procedure RP 202 "Radiation Work Permits", Revision 1 dated February 19, 1988, and several radiation work permits in effect. The inspector concludes that the RWP forms in use are satisfactorily organized to facilitate worker comprehension. This item is closed.

(Closed) Unresolved Item 50-272/87-19-01 and 50-311/87-21-01; Data collection concerns for pump and valve testing. The inspector reviewed the Operations Newsletter dated April 8, 1988, in which the operators were directed to document and maintain all data resulting from pump and valve surveillance tests regardless of the acceptability of the data and reasons therefore. The inspectors will continue to monitor licensee actions in this regard during routine inspections. This item is closed.

3. Operational Safety Verification (71707, 71709, 71881)

3.1 Inspection Activities

On a daily basis throughout the report period, inspections were conducted to verify that the facility was operated safely and in conformance with regulatory requirements. The licensee's management control system was evaluated by direct observation of activities, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety system status and limiting conditions for operation, and review of facility records. The licensee's compliance with the radiological protection and security programs was also verified on a periodic basis. These inspection activities were conducted in accordance with NRC inspection procedures 71707, 71709 and 71881 and included weekend and backshift inspections.

3.2 Inspection Findings and Significant Plant Events (93701)

3.2.1 Unit 1

Unit 1 operated at 100% power throughout the inspection period.

3.2.2 Unit 2

Unit 2 began the report period operating at 100% power.

Based upon the continuing Salem electrical distribution system design review, the licensee concluded on May 9, 1988, that 37 electrical circuits did not have adequate containment penetration conductor overcurrent backup protection. Of the 96 circuits specified in Unit 2 Technical Specification (T.S.) 3.8.3.1, 37 circuits had marginal or unacceptable coordination between the backup circuit breaker and the conax connector at the containment penetration. Under certain circumstances, the rating of the conax connector could have been reached prior to the backup breaker tripping open if a circuit fault were to develop and the primary breaker failed to trip. In all cases, the primary circuit breakers (located between the backup breaker and the conax connector) had proper coordination and were operable. The purpose of the primary and backup containment penetration conductor overcurrent protective devices is to ensure that containment electrical penetrations and containment integrity will not be adversely affected by a circuit failure inside of containment.

The licensee entered T.S. action statement 3.8.3.1a which required the affected circuits to be de-energized by tripping the backup circuit breakers within 72 hours. A review of the operational impact on completing the action statement for each of the 37 circuits identified 2 concerns. First, there were 5 circuits (21 and 22 containment sump pumps, RHR valves 21SJ44 and 22SJ44, and one reactor coolant drain tank pump) that were essential to continued plant operations. Secondly, because the backup breakers are frequently main feeders to load centers, complying with the action statement would have had an adverse affect throughout the plant.

The first issue was resolved by implementing major design change 2SC-2001 "Backup Protection for Electrical Penetrations" which installed in-line molded case circuit breakers with acceptable trip ratings in each of the five circuits. The inspector found this design change and related engineering evaluation (S-1-ZZ-XX-EEE-240-0 - "Penetration Circuit Protection Analysis") to be acceptable. To address the second issue, the licensee submitted an emergency Technical Specification change request on May 10, 1988. This submittal proposed that with one or more of the containment penetration conductor overcurrent protective devices inoperable, the affected circuit be de-energized by tripping "either" the primary or backup protective device. The submittal reasoned that opening either protective device will completely de-energize that portion of the circuit passing through the containment penetration. After review of the licensee's submittal and conference calls on May 10, and May 12, between the licensee, NRR, and Region I, a temporary waiver of compliance to change action statement T.S. 3.8.3.1a was issued by NRR on May 12, 1988. The licensee verified compliance with the new action statement and initiated a weekly check of open primary breakers that same day. In the request for amendment, dated May 10, 1988, the licensee committed to repair all of the remaining circuits prior to startup from the September 1988 refueling outage. (IFI 311/88-13-01) Salem Unit 1 does not have T.S. requirements on electrical containment penetration protection, however this issue will be reviewed on Unit 1 after the Unit 2 review is completed. (IFI 272/88-13-01)

On May 13, 1988, the Salem Unit 2 control room operator was inserting control rods (Bank D) for a turbine valve test load reduction. At approximately 213 steps, the reactor tripped on power range high negative neutron flux rate. All plant systems responded to the trip as designed. Safety parameter display system (SPDS) data indicated that control rod 1D3 dropped during rod insertion. Investigation and testing of the rod control system by the licensee did not reveal any equipment deficiencies that could have caused the trip. However, a 16 volt power supply in the rod control logic cabinet and a power cabinet alarm circuitry card were replaced as precautionary measures. The inspector reviewed the SPDS data, the licensee's test data and results, and Licensee Event Report 88-009-00 and agrees with the licensee's conclusions that control rod 1D3 dropped during rod insertion and that the test results are inconclusive as to the cause of the dropped rod and subsequent trip. The reactor was returned to criticality on May 15, 1988. The unit remained at 100% power throughout the rest of the report period. No further problems with the rod control system were experienced.

3.3.3 Both Units

On May 19, 1988, the licensee identified non-seismically qualified pneumatic-electric (PE) relays associated with the emergency diesel generator (EDG) fire protection equipment. The licensee notified the NRC of the deficiencies via the Emergency Notification System (ENS) and informed the resident inspectors. The deficiencies were identified during the detailed design phase for a modification required for previously identified fire protection system problems. The mercury contacts associated with the PE relays interface with Class 1E circuits for the fans in the diesel generator area. In a seismic event the mercoid relays in question could chatter and prevent the EDG room exhaust fans from operating. Diesel generator operability is contingent upon diesel generator area ventilation to cool the generator. Previous analyses indicate the EDG's can remain operable without ventilation for at least 20 minutes. SORC reviewed the deficiency and concluded that diesel and diesel ventilation operability was ensured by the fact that operations and/or fire protection operators were available to reset the PE relays. Fire protection personnel routinely perform a surveillance procedure which delineates, in part, how to reset the relays. SORC's immediate short term corrective actions directed that specific relay resetting instructions be provided for the operations department. The inspector reviewed the implementation of the SORC directive. Operations supervision was given a copy of the fire protection surveillance procedure and told which steps to follow to reset the relay. Night shift operators received training (on resetting the relay) prior to assuming their watch, but day shift operators did not receive training until several hours into their shift. In addition, operations supervision was unclear as to whether operators or fire protection personnel had primary responsibility for resetting the relay in a seismic event. The inspector requested that the licensee demonstrate that the relay could be reset by an operator within the 20 minute time frame, using the fire protection surveillance procedure and prior to formal training. This was accomplished successfully. However, the inspector was concerned with the informality of implementation of the corrective actions intended by SORC. It appears that improvements are warranted in this area to ensure accurate and timely implementation of immediate short term corrective actions. This issue will be reveiwed further during subsequent routine inspections. Design Change Package (DCP) ISC/2SC-1609 "Modification of the Fire Protection System for the Diesel Generator and Control Areas to Comply with 10CFR50 Appendix R" will eliminate the PE relay seismic concerns. These DCP's are scheduled for installation during June, 1988.

No violations were identified.

4. Maintenance Observations (62703)

The inspector reviewed the following safety related maintenance activities to verify that the activities were conducted in accordance with approved procedures, Technical Specifications, NRC regulations, and industry codes and standards.

Work Order Number	Description
880509128	Troubleshoot No. 21 Waste Gas Compressor controls and valves

The inspector observed poor radiological controls and housekeeping demonstrated by workers performing this work order. Tools, cement chips, and used anti-contamination clothing were strewn across the contaminated area boundary. The inspectors concerns were brought to the attention of radiation protection supervision and corrected.

Work Order	Design Change Package	Description
870903044	2SM0362	Installation of Thermocouples (4) on 23 Auxiliary Feedwater Pump discharge lines for backleakage monitoring.
Various	2SC-2001	Installation of backup protection (breakers) for containment electrical penetration (21 and 22SJ44's, 22 reactor coolant drain tank pump, 21 and 22 containment sump pumps).

The inspector witnessed portions of installation and testing associated with these DCP's and found the activities to be acceptable. No violations were identified.

5. Surveillance Observations (61726)

5.1 Inspection Activity

During this inspection period, the inspector performed detailed technical procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspector verified that the surveillances were performed in accordance with Technical Specifications, licensee approved procedures, and NRC regulations. These inspection activities were conducted in accordance with NRC inspection procedure 61726.

The following surveillance tests were reviewed, with portions witnessed by the inspector:

2PD-16.2.006	Source Range At Power Channel Functional Test
PI/S-AF-3	Auxiliary Feed Water Backleakage
OP-TEMP-8805-2	Stroke Test Valve 22SJ44 following installation of backup breaker.
M2B	Fuel Handling Crane Periodic Inspection and Operational Tests
2IC-2.6.020	Pressurizer Level Transmitter 2LT-459 Functional Test

The inspector concluded that these surveillance tests were properly conducted.

No violations were identified.

6. Engineered Safety Feature (ESF) System Walkdown (71710)

6.1 Inspection Activity

The inspectors independently verified the operability of selected ESF systems by performing a walkdown of accessible portions of the system to confirm that system lineup procedures match plant drawings and the as-built configuration. The ESF system walkdown was also conducted to identify equipment conditions that might degrade performance, to determine that instrumentation is calibrated and functioning, and to verify that valves are properly positioned and locked as appropriate. This inspection was conducted in accordance with NRC inspection procedure 71710. The Units 1 and 2 Intermediate Head Safety Injection (SI) Systems were inspected. The inspector noted boric acid crystals built up around the stems of SI pump mini-flow isolation valves 1SJ67, 2SJ67 and 2SJ68. These conditions were brought to the attention of operations shift supervision who directed the valve stems to be cleaned off. These valves are normally open with the power to the Limitorque motor operator locked out. The valves must be closed before the SI pumps can be fed from the residual heat removal system (RHR) during the recirculation phases of emergency core cooling. The inspector reviewed the previous inservice testing procedure 4.0.5 V-SJ-5 completed for each valve (done in Mode 5 prior to startup) and verified that the valves stroked within the required time (10 seconds). Overall system conditions were found to be acceptable.

No violations were identified.

7. Review of Licensee Reports (90712, 90713, 92700)

Upon receipt, the inspector reviewed licensee event reports (LERs) as well as other periodic and special reports submitted by the licensee. The reports were reviewed for accuracy and timely submission. Additional followup performed at the discretion of the inspector to verify corrective action implementation and adequacy is detailed with the applicable report summary. The following reports were received and reviewed during the inspection period:

- Unit 1 Monthly Operating Report - April, 1988
- Unit 2 Monthly Operating Report - April, 1988

- Unit 1 Special Report 88-2 Fire Barriers Impaired For Greater Than Seven (7) Days

On April 29, 1988, as part of an 18 month surveillance and in conjunction with a licensee initiated Penetration Seal Review Program, Fifty-two (52) fire barrier electrical and mechanical penetrations (of approximately 6000 total penetrations) were found missing or degraded. Hourly fire watches had already been established in the applicable areas for previous fire protection deficiencies. PSE&G letter NLR-N88037 dated March 4, 1988 to the NRC delineates the licensee's schedule for completion of the Penetration Seal Review Program including repair of degraded seals. The inspector had no further questions at this time.

- Unit 1 LER Supplement 87-018-01 Improperly Calibrated Lead/Lag and Derivative Amplifiers

This event was discussed in combined NRC Inspection Report 50-272/311/87-36. The licensee identified an error in calibrating the dynamic response (lead/lag and derivative functions) of certain reactor protection circuits. The licensee concluded that the miscalibration of lead/lag circuits resulted in conservative tripping of the RPS circuits such as low steam line pressure and low pressurizer pressure. Further evaluation was necessary to determine the effect of the improperly calibrated derivative amplifiers. On May 25, 1988, the licensee supplemented the original LER concluding that the overpower delta-T (OPdT) trip function would have tripped non-conservatively only when reactor coolant system (RCS) temperature was increasing rapidly. The allowable trip setpoint would not be exceeded unless the RCS temperature rose at a rate greater than 9 degrees F per minute. Further, the licensee noted that OPdT protection is a backup for the high neutron flux protection function and it also limits the required range for overtemperature delta T protection. These functions are not taken credit for in the Salem accident analyses.

NRC review of this event recognized the inoperability of the OPdT function under certain circumstances. However, the licensee identified the problem and promptly corrected and reported the condition. No further corrective actions were identified. In consideration of the low safety significance as delineated above and to encourage further improvement in dynamic testing of RPS functions, NRC determined that no violation would be cited in accordance with 10CFR 2, Appendix C (272/311/88-13-02).

- Unit 1 LER Supplement 88-001-01 Diesel Generator Day Tanks Seismic Deficiency

This event was discussed in combined inspection 272/311/88-03. Licensee investigation has attributed the root cause of the lack of anchoring of the Unit 1 day tanks to inadequate design and design review, in that anchoring was not specified in historical design records. PSE&G civil drawings have been updated to address anchor requirements. The inspector had no further questions at this time.

- Unit 2 LER 88-006-00 Reactor Trip/False No. 23 Reactor Coolant Loop Low Flow Signal

This event was discussed in combined inspection 272/311/88-11 and the inspector had no further questions following review of the LER.

- Unit 2 LER 88-007-00 Reactor Trip Resulting From Faulty Turbine Electro-hydraulic Controls (EHC) Response

This event was reviewed in combined inspection 272/311/88-11. The inspector had no further questions following review of this LER.

- Unit 2 LER 88-008-00 Fire Protection Containment Isolation Valve Missed Surveillance

On April 26, 1988, the licensee identified that valve 2FP147 had not been surveilled within the Technical Specification required interval of every 92 days. This valve is normally closed and fails closed. It is manually opened when deluge flow to the reactor coolant pump is required for fire suppression. The 2FP147 valve was subsequently tested satisfactorily. At the time the surveillance was scheduled, the licensee decided to postpone the surveillance on 2FP147 and do it later in the procedure along with check valve 2FP148 since testing of both valves requires manual isolation of the containment fire protection header to preclude inadvertent deluge of the reactor coolant pumps. When 2FP148 was tested and since the test data for the two valves is recorded on different checkoff sheets, testing on 2FP147 was forgotten. Supervisory review of the completed surveillance procedure failed to identify the missing data. The applicable surveillance procedure SP(O)4.0.5 V-MISC has been revised to perform surveillance of 2FP147 and 2FP148 at the same time. Corrective disciplinary action has been taken by the licensee for those personnel involved and the need to maintain attention to detail was reemphasized with operations personnel. The inspector has determined that this event is similar in nature to one for which a violation was previously cited (311/87-18-01) in that personnel performed and management reviewed the completed procedure, failing to identify the incomplete checkoff sheet and resulting in a portion of the surveillance not being accomplished within the required time. This represents a licensee identified violation of Technical Specifications 4.0.5 and 6.8.1c. (311/88-13-03) Since appropriate corrective actions have already been taken by the licensee, no further action is required.

The following events are discussed in Section 3.2 of this report.

- Unit 1 LER 88-010-00 Diesel Generator Cardox PE Relay Not Seismically Qualified-Potential Loss of D/G Ventilation

- Unit 2 LER 88-009-00 Reactor Trip/Dropped Control Rod

8. Exit Interview (30703)

The inspectors met with Mr. J. Zupko and other licensee personnel periodically and at the end of the inspection report to summarize the scope and findings of their inspection activities.

Based on Region I review and discussions with the licensee, it was determined that this report does not contain information subject to 10 CFR 2 restrictions.