U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Report No:	50-354/98-09
Licensee:	Public Service Electric and Gas Company
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, New Jersey 08038
Dates:	August 17 - 26 and September 8, 1998
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EXECUTIVE SUMMARY

Hope Creek Nuclear Generating Station
NRC Inspection Report 50-354/98-09

During the period of August 17 - 28, 1998 and on September 8, 1998, the NRC conducted an engineering inspection at the Hope Creek Nuclear Generating Station. The objectives of the inspection was to assess: (1) the current fuel reload core design and effectiveness of the engineering functions in this area; (2) the current level of engineering backlogs and ongoing efforts to reduce it; and (3) the adequacy of the actions to address previously identified issues. During the inspection periods, the plant remained at or near full power.

Maintenance

- The inspectors' review of the LPRM and APRM calibration process identified no areas of concern with the procedures or their implementation. (Section M1.1)

Engineering

- The difference between the PANACEA core design code and the P-1 on-line core monitoring program predictions indicated the need for a review by GE of the applicability of the P-1 calculation to the new fuel designs. However, the approach taken by the licensee to account for the observed differences between the PANACEA and the P-1 prediction was acceptable and resulted in conservative local fuel limit values. This licensee's approach, however, could unnecessarily restrict the Hope Creek plant operation. (Section E1.1)

- The licensee had not developed a procedure for preparing the FRED and OPL-3 data transfer documents, indicating an insufficient level of control for this safety-related activity. Also, the review of the Cycle-8 FRED and OPL-3 data transmittal indicated that the data had not been independently reviewed and verified. However, no specific concerns were identified with this issue. (Section E1.2)

- PSE&G's thorough review and confirmatory analysis of the spacer deviation evaluation demonstrated a high level of awareness and quality assurance in reload safety analyses. When the vendor supporting test data was unavailable, the initial approach taken by PSE&G to account for the spacer deviation was very conservative. (Section E1.3)

- The licensee's root cause analysis, development of comprehensive corrective actions, and proactive followup of GE reported errors in the Hope Creek Safety Limit Minimum Critical Power Ratio (SLMCPR) calculations and quality control problems with the manufacturing of fuel pellets for the Hope Creek reload bundles, demonstrated the licensee commitment to safety and the ability to provide proper vendor oversight during emerging issues. (Section E1.4)

- The reactor engineering staff interviewed had a good understanding of the 50.59 process for addressing design and license changes and the process was being properly applied. Also, they maintain complete reload analysis documentation and management provided effective licensing oversight. (Section E1.5)

- The amount of backlogged engineering activities continued to be high, but improving in all areas, albeit not at the rate previously planned by management. Steps had been taken to correct the trend, indicating that management was committed to reduce the backlog to within their own stated limits. Backlogged activities had been properly prioritized and none of the sample PIRs reviewed indicated the need for immediate corrective action. The progress made in backlog reduction by the valve engineering group was slower than that made by other groups. No concerns were, nonetheless, identified with the prioritization and safety impact of the sample items reviewed. (Section E2.1)

- Except for minor discrepancies, no examples were identified of inadequate performance by the current staff. Training of new staff and ongoing training was acceptable. Participation in industry groups was proactive. (Section E5.1)

- Installation of the DCP satisfied the NRC concerns regarding interaction between the fire suppression and the EDG room ventilation systems. Also the carbon dioxide test anomalies were being properly addressed and no concerns existed with the fire spreading beyond established boundaries. (Section E8.1)

- The evaluation of the relays safety functions and the revision of the relays service life calculation were acceptable. (Section E8.3)

- The licensee had properly addressed the breaker failure event; the root cause analysis had accurately evaluated the breaker failure modes; and the corrective actions were appropriate and properly managed. (Section E8.4)

- The licensee's evaluation of the causes of three examples of inadequate test control and the actions to resolve the issues were acceptable. (Section E8.5)

- The licensee's evaluation and resolution of six design control issues were acceptable. (Sections E8.6)

- The licensee's investigation of the fire protection program deficiencies was thorough and the resulting corrective actions appropriate. (Section E8.8)

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Report Details

During the period of August 17 - 26, 1998, and on September 8, 1998, the NRC conducted an engineering inspection at the Hope Creek Nuclear Generating Station. The objectives of the inspection was to assess: (1) the current fuel reload core design and effectiveness of the engineering functions in this area; (2) the current level of engineering backlog and ongoing efforts to reduce it; and (3) the adequacy of the actions to address previously identified issues. During the inspection periods, the plant remained at or near full power.

II. Maintenance

M1 Conduct of Maintenance

M1.1 LPRM/APRM Instruments Calibration

a. Inspection Scope

The local power range monitors (LPRMs) provide local neutron flux signals at various incore locations. The assemblies include a hollow dry tube for the traversing incore probe (TIP) system. The TIP system provides the means for measuring the neutron flux and calibrating the LPRM detectors. The LPRM detectors provide their input to the Average Power Range Monitor (APRM) and other indication and monitoring systems. The APRM is used for computing core average flux, providing protective trips and, therefore, preserving the integrity of the fuel cladding.

Periodic calibration of the LPRMs is necessary to ensure that the LPRM flux amplifier outputs provide an accurate indication of local power conditions. Calibration of the APRM is necessary to ensure that the APRM channel output provides an accurate indication of reactor core thermal power. The inspectors reviewed the Hope Creek nuclear instrumentation calibration procedures and the frequency of calibration.

b. Observations and Findings

At Hope Creek, calibrations of the LPRM are performed at least once every 1000 effective hours, using the Traversing Incore Probe (TIP) system. For the calibration of the LPRM system the licensee developed two procedures, HC.RE-ST.SE-0003(Q), "LPRM Calibration," and HC.IC-CC.SE-0029(Q), "Channel Calibration Nuclear Instrumentation System LPRM Gain Calibration." The inspectors' review of selected sections of these procedures identified no areas of concern. The inspectors also determined that the current revisions had undergone the required review process and had received the appropriate 50.59 applicability review.

For the calibration of the APRM the licensee uses Procedure HC.RE-ST.SE-002(Q), APRM Calibration Surveillance." The APRM channel output is required to be assessed by the licensee at a minimum frequency of once per week during power operation. In addition the surveillance is performed under reactor startup conditions as necessary to maintain proper APRM calibration. The inspectors reviewed appropriate sections of the procedure and found it acceptable. As in the case of the above procedures, the APRM procedure had undergone the required reviews and changes from the previous revision had been evaluated under 10 CFR 50.59. The use of the procedures was discussed with a member of the operating staff as well as a reactor engineer. The inspectors identified no concerns with the procedures or their application.

c. Conclusions

The inspectors' review of the LPRM and APRM calibration process identified no areas of concern with the procedures or their implementation.

III. Engineering

E1 Conduct of Engineering

E1.1 Adjustment of the PANACEA Reload Design Analysis Code

a. Inspection Scope

As part of the evaluation of the PSE&G reload analysis process, the inspectors reviewed the Hope Creek Cycle-8 reload analysis documentation. The review included the PSE&G Cycle-8 Fuel Cycle Analysis Request, dated March 4, 1996, and supporting documentation.

b. Observations and Findings

The Fuel Cycle Analysis Request (FCAR) Document is provided by PSE&G to the fuel vendor and specifies the fuel contract requirements, fuel cycle technical basis, cycle energy utilization and operation plans, and special cycle-specific analysis instructions. Under special instructions, the Cycle-8 FCAR provided comparisons of the General Electric (GE) PANACEA and the P-1 process computer predictions of the local fuel limits (linear heat generation rate, MAPRAT, and maximum fraction of limiting power density). These comparisons are used to adjust the PANACEA core design code to ensure agreement with the P-1 on-line core monitoring predictions. Also, the PANACEA steady-state physics code is used by GE to perform the cycle-specific reload core design analysis.

The inspectors review of the PANACEA/P-1 comparisons included in the FCAR determined that PANACEA under-predicted the local fuel limits for Cycles 4 through 6 by as much as 15% relative to P-1. This difference was well outside the expected uncertainties of both of these codes. It was not clear which calculation was providing the most accurate prediction; however, the inspectors noted that: (1) several substantial fuel design changes had been introduced since the original bench marking and verification of the P-1 computer; and (2) Hope Creek was the only BWR plant presently using P-1 for on-line surveillance. Therefore, a review by GE of the applicability of the P-1 calculation to the new fuel designs may be necessary to provide additional verification of the P-1 predictions.

The NRC review, in 1991, of the PANACEA/P-1 differences concluded (IR 50-354/91-21) that: "The licensee was aggressive in the analysis of predictive problems with GE-9 fuel and their effects on cycle 4 core performances. Actions taken to maintain adequate margins to core thermal limits were conservative and appropriate."

As for Cycle-5 through Cycle-7, the Cycle-8 FCAR requires that additional margin be included in the PANACEA results to account for the PANACEA under-prediction of the local fuel limits. Also, because it is not known whether PANACEA or P-1 provides the most accurate predictions, PSE&G has conservatively assumed that the maximum values calculated by P-1 are correct and has accordingly reduced the thermal margin calculated by PANACEA. However, more recent comparisons indicated that the P-1 predictions were decreasing relative to PANACEA. In this case, if this trend continued the PANACEA predictions could become limiting and the licensee should reevaluate the use of the P-1 values. The licensee recognized that the conservative approach taken with respect to PANACEA and P-1 predictions could unnecessarily restrict the Hope Creek plant operation.

c. Conclusions

The difference between the PANACEA core design code and the P-1 on-line core monitoring program predictions indicated the need for a review by GE of the applicability of the P-1 calculation to the new fuel designs. However, the approach taken by the licensee to account for the observed differences between the PANACEA and the P-1 prediction was acceptable and resulted in conservative local fuel limit values. The inspectors also concluded that the licensee's approach could unnecessarily restrict the Hope Creek plant operation.

E1.2 Preparation of the Fuel Release and Engineering Data (FRED) and the Operating Plant Parameters (OPL-3) Data

a. Inspection Scope

The inspectors reviewed the licensee/vendor interface, including the transfer of safety-related data from PSE&G to the fuel vendor. Specifically, the review focused on the transfer of the Fuel Release and Engineering Data (FRED) and the Operating Plant Parameters (OPL-3) for Hope Creek Reload-7 (Cycle-8) to the fuel vendor. The focus was on the reliability of this data and the conformance to PSE&G procedures.

To identify the critical licensee/vendor data transfers, the inspectors reviewed the Hope Creek Cycle-8 reload analysis file, including the Fuel Design Request and the 10 CFR 50.59 Safety Evaluation. The inspectors also reviewed the cycle-specific analyses including: (1) loose parts evaluations; (2) analysis of as-built deviations in the upper tie plate geometry; and (3) evaluation of the spacer band tab deviations.

b. Observations and Findings

The inspectors noted that both the FRED and OPL-3 data transfers included important and extensive listings of plant design and operating data that are used in the fuel vendor safety analysis. For example, these reports listed plant operating conditions (e.g., pressure, flow, and temperature), operating improvement options, technical specification requirements, and transient input data.

The inspectors observed that, while detailed procedures existed for: (1) performing design analyses (ND.NF-AP.ZZ-0005(Q)); (2) controlling computer software (ND.NF-CP.ZZ-0003(Q)); and (3) performing 10 CFR 50.59 Safety Evaluations (NC.NA-AP.ZZ-0059(Q)); the licensee had not developed procedures for controlling the preparation of the FRED and OPL-3 data. Because the FRED and OPL-3 data was used in important safety applications, the inspectors considered the lack of procedures for preparing these documents incompatible with the criteria used for other activities and insufficient for assuring the accuracy of the data developed. The licensee initiated action to prepare a procedure for this activity.

The inspectors' review of the Cycle-8 FRED and OPL-3 data transmittals to GE also noted that, while the data had been discussed with GE and the cover letters had been properly signed, the licensee had no record of having independently reviewed and verified the accuracy of the attached data. This was inconsistent with Section 5.2.1 of PSE&G Procedure ND.NF-AP.ZZ-0003(Q), Revision 0, "Nuclear Fuel Section Correspondence Control," which requires that outgoing data be reviewed by Nuclear Fuels Engineers. However, discussions with the PSE&G technical staff and a detailed review of the data by the inspectors identified no specific discrepancies. The lack of procedure for developing safety-related data and the licensee's failure to conduct independent reviews of such data are minor violations of Appendix B, Criterion V, not subject to formal enforcement action.

c. Conclusions

The licensee had not developed a procedure for preparing the FRED and OPL-3 data transfer documents, indicating an insufficient level of control for this safety-related activity. Also, the review of the Cycle-8 FRED and OPL-3 data transmittal indicated that the data had not been independently reviewed and verified. However, no specific concerns were identified with this issue.

E1.3 PSE&G Evaluation of Fuel Spacer Band Flow Tabs Outside of Design Specifications

a. Inspection Scope

The inspectors reviewed the PSE&G evaluation of the effect of the spacer band flow tabs, determined by GE to be outside the design specification, to assess their ability to carry out cycle-specific reload design analyses.

b. Observations and Findings

Prior to the startup of Cycle-8, General Electric informed PSE&G that the spacer flow tabs, which direct flow back into the fuel bundle to improve critical power performance, were not oriented within the angle tolerance limits specified in the design drawings. GE also stated that the spacer tab deviation was not a safety concern and, based on available test data, did not impact the critical power performance of the fuel bundles.

PSE&G attempted, but was unable to independently confirm GE's conclusions regarding the spacer tab deviation, because the GE test data was not available. Therefore, they conservatively took no credit for the spacer band flow tabs and in the 10 CFR 50.59 Safety Evaluation applied a 6% ΔCPR penalty to the Cycle-8 MCPR operating limit. Subsequently, prior to Cycle-8 startup, PSE&G obtained the test data, independently confirmed that the spacer tab deviation had no impact on the bundle critical power, and removed the 6% ΔCPR penalty they had applied to the Cycle-8 operation. The inspectors noted that the spacer band tab analysis was described in the reload documentation and had been independently verified.

c. Conclusions

PSE&G's thorough review and confirmatory analysis of the spacer deviation evaluation demonstrated a high level of awareness and quality assurance in reload safety analyses. The inspectors also concluded that, when the vendor supporting test data was unavailable, the initial approach taken by PSE&G to account for the spacer deviation was very conservative.

E1.4 Licensee Response to General Electric (GE) Error Reports

a. Inspection Scope

As part of the evaluation of the Hope Creek reload safety analyses process, the team reviewed the licensee response to reported errors in Safety Limit Minimum Critical Power Ratio (SLMCPR) calculations for Hope Creek and to the discovery of low density pellets in Hope Creek reload bundles.

b. Observations and Findings

GE reported errors in Safety Limit Minimum Critical Power Ratio (SLMCPR) calculations for Hope Creek and the discovery of low density pellets in Hope Creek reload bundles. The inspectors found the licensee's evaluation of these two unrelated problems, one an analytical error and one a manufacturing quality control error, to be thorough and comprehensive. To address these issues, the licensee developed a detailed recovery plan with well defined milestones. The inspectors also found evidence of good licensee followup activity of the GE corrective actions associated with these issues and that the overall emerging issue oversight, root cause evaluation, and corrective action followup had been strong in past cycles. Lastly, the inspectors observed a high number of recent manufacturing audits, to review the vendor recovery plan and to verify resolution of past problems.

c. Conclusions

The licensee's root cause analysis, development of comprehensive corrective actions, and proactive followup of GE reported errors in the Hope Creek Safety Limit Minimum Critical Power Ratio (SLMCPR) calculations and quality control problems with the manufacturing of fuel pellets for the Hope Creek reload bundles, demonstrated the licensee commitment to safety and the ability to provide proper vendor oversight during emerging issues.

E1.5 Hope Creek Licensing Evaluations and Reload Safety Documentation

a. Inspection Scope

As part of the evaluation of the Hope Creek reload safety analyses process, the inspectors reviewed the overall quality of licensing oversight and documentation. Specifically, they reviewed the quality of the 10 CFR 50.59 process and license change request process by reviewing applicable procedures and documents, including internal memoranda and correspondence with the vendor. They also conducted interviews of the Fuel Engineering and Licensing staff.

b. Observations and Findings

The inspectors found the safety evaluation process and the process for requesting license changes to be acceptable. Further, the senior Fuels staff, the Licensing Group and the Reactor Engineering staff demonstrated a good understanding of the 10 CFR 50.59 requirements and of the mechanism for requesting license changes and Technical Specification updates. The inspectors evaluated the licensee's understanding of the GE Supplemental Reload Licensing Report (SRLR) and its relationship to the licensee-generated Hope Creek Core Operating Limits Report (COLR). They found the senior staff members to be knowledgeable in the matter.

c. Conclusions

The inspectors concluded that the staff interviewed had a good understanding of the 50.59 process for addressing design and license changes and that the process was being properly applied. The inspectors also concluded that they maintain complete reload analysis documentation and that management provided effective licensing oversight.

E2 Engineering Support of Facilities and Equipment

E2.1 Management of Engineering Backlog

a. Inspection Scope (37550)

The NRC reviewed the engineering backlog previously, in conjunction with the restart of the Salem Units (IR 50-272; 311/97-18 and 97-21). That review concentrated on the content of the volume of backlogged activities related to the

Salem Units only. At that time, the NRC concluded that the backlog was extensive, but properly managed and prioritized.

The purpose of the current review was to evaluate the current volume of backlogged engineering activities and the criteria used by PSE&G to prioritize such activities. The review also sampled the list of Hope Creek backlogged work to ensure that the items had been properly classified and did not involve work that, if delayed, might impact the safe operation of the plant.

b. Observations and Findings

The inspectors' review of backlogged engineering activities determined that, at the time of the inspection, the quantity of open Performance Improvement Requests (PIRs) and open Design Change Packages (DCPs) for both Salem and Hope Creek totaled approximately 6500 and 130, respectively. Earlier in the year the licensee had planned, at least in the Design Engineering area, to achieve a work backlog equivalent to six months by February 1999. This date was selected to ease the burden of the Hope Creek and Salem Units refueling outages scheduled to begin with Hope Creek, in February 1999.

The inspectors' review of graphs provided by the licensee indicated that some progress had been made, since the last review, in backlog reduction. However, the decrease was smaller than anticipated. For instance in the design area the actual burn off rate was approximately half the amount expected. The licensee has already taken steps to assign some activities to contractor personnel. Hope Creek-specific graphs indicated similar results.

Previous NRC inspection had not specifically addressed System and Maintenance Engineering backlogs, because work activities were viewed as event driven and, therefore, ongoing. For Hope Creek, System Engineering reported, on the average, approximately 550-600 items in the corrective action program. Of these, approximately 60 were overdue. In the Maintenance Engineering area, the amount of open activities was fluctuating around 400. Of these, approximately 100 were more than 120 days old.

In the Valve Engineering area, the inspectors observed that the average amount of open activities was approximately 375, with a slight downward slope during the last three months. Of the above activities, more than 1/3 were more than 120 days old. The licensee indicated that current trends yielded an average backlog reduction of one item per week, but planned the addition of one engineer to the group. Despite the addition of an engineer, projections to the end of 1998 indicated a backlog of approximately 325 items, indicating a slower progress than in other engineering groups.

To address the content of the backlog activities, the inspectors selected approximately one hundred PIRs from the design, system and maintenance engineering groups, concentrating primarily on those activities that were at least one year old and indicated the potential for being safety-related. The review of the selected PIRs identified no issues that required immediate licensee's attention.

c. Conclusions

The inspectors concluded that the amount of backlogged engineering activities continued to be high, but improving in all areas, albeit not at the rate previously planned by management. Steps had been taken to correct the trend, indicating that management was committed to reduce the backlog to within their own stated limits. The inspectors also concluded that the activities had been properly prioritized and that none of the sample PIRs reviewed indicated the need for immediate corrective action. The backlog reduction progress of the valve engineering group was slower than the other groups. No concerns were, nonetheless, identified with the prioritization and safety impact of the sample items reviewed.

E5 Engineering Staff Training and Qualification

E5.1 Staffing, Training and Qualification

a. Inspection Scope

The inspectors reviewed the current staffing in terms of numbers, experience and qualifications, and training requirements.

b. Observations and Findings

Due to recent loss of personnel and organizational changes, the current Fuels group includes a number of new members, although some of them, including the group supervisor, had previous experience in Fuels and Reactor Engineering functions. The permanent PSE&G staff was currently augmented by four experienced consultants, one with extensive Hope Creek experience. Two slots were open for experienced engineers and the licensee was actively seeking candidates and planned continued use of consultants to fill staffing gaps and to provide training for the transition period.

The inspectors' review of the staff qualifications also determined that there were only two Station Qualified Reviewers (SQRs), but two of the consultants are also SQR certified and the licensee planned to have two additional SQRs by year end. The current training procedures, along with job-specific qualification guides for both technical and supervisory staff, were adequate in scope and depth. Further, the current SQRs training qualifications were up-to-date. The inspectors also observed proactive participation in NUPIC sponsored audits, and involvement with the BWR Owners Group (BWROG) Reload Analysis and Core Management Committee (RACMC) Vendor Oversight program initiatives.

c. Conclusions

Except for minor discrepancies, the inspectors identified no examples of inadequate performance by the current staff. Training of new staff and ongoing training was acceptable. Participation in industry groups was proactive.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Violation 50-354/96-09-03: De-Facto Modification without Safety Evaluation

On October 29, 1996, the NRC determined that PSE&G had conducted de facto changes to the facility as described in the UFSAR without a written safety evaluation which provided the bases for the determination that the changes did not constitute an unreviewed safety question. The changes involved: (1) the incorrect use of isolation devices (relays) between safety and nonsafety-related circuits in the emergency diesel generator (EDG) room ventilation system, and (2) the use of nonsafety-related fire suppression devices to close the EDG room fire dampers and shutdown the associated ventilation system.

In their response to the notice of violation, letter No. LR-N96436, dated January 9, 1997, PSE&G stated that a temporary modification had been implemented to disconnect four nonsafety-related relays in the ventilation system and that various options were under consideration for permanent resolution of the circuit interface concern. The NRC found the response insufficient to address the inadvertent actuation concern and in a letter, dated June 10, 1997, requested that the permanent resolution address the NRC concern. In a subsequent letter, dated October 21, 1997, the licensee informed the NRC that they had initiated a design change package (DCP), No. 4EC-3644, to permanently eliminate the interaction between the fire suppression and the EDG ventilation systems. This change package entailed the elimination of the fire dampers and an appropriate increase of carbon dioxide discharge to account for the associate protection volume increase.

Prior to the final installation of the plant modification, PSE&G conducted carbon dioxide discharge tests to confirm the adequacy of its concentration, but they experienced two test malfunctions, one involving the premature reclosure of the discharge header isolation valve, due to failed seals, and the other the blowing open of the EDG room fire doors, due to excessive room pressurization. The NRC review of the modification package and of the discharge test results was documented in Inspection Report 50-354-97-07. Installation of the DCP, completed during the November 1997 refueling outage satisfied the NRC concerns regarding interaction between the fire suppression and the EDG room ventilation systems. This item is closed.

Regarding the test malfunctions, the licensee addressed them in two condition resolutions (CRs), 970909107 (Failed CO_2 Dump Test) and 970916281 (Fire Door Failure During CO_2 Discharge), dated September 8 and September 18, 1997, respectively. The licensee's evaluation of the failures and related NRC concerns were extensive. Specifically, regarding the failure of the header discharge valve seals, the licensee concluded that, in the event of a fire in any of the nine CO_2

protected areas, the fire damage might have been more extensive, but of limited safety significance, because of the other fire protection feature designed into the system, including barriers and manual fire fighting capabilities. Corrective actions that were initiated to prevent similar future failures involved the review of required valve maintenance and the inclusion of the CO_2 system valves in the preventive maintenance program.

Regarding the increase in room pressure that resulted in the blowing open of the EDG fire doors, the licensee once again determined that the impact on safety would have been minimal and that fire would have not spread to other areas of the plant. They attributed the event to lack of knowledge of the pressure at which the doors would blow open, complicated by overconservative calculations of the needed CO_2 discharge to achieve concentration, insufficient consideration for heating effect from the ventilation system, and lack of leakage pathways.

The CO_2 remained in the manual mode while the licensee evaluates alternatives to reduce excessive pressure buildup in the EDG rooms and to prevent fire doors from blowing open. The alternatives being considered were delineated in the evaluations and the required actions included in the licensee corrective action program. Resolution was currently scheduled for February 25, 1999. Based on their review of the CRs and associated evaluations, as well as their discussions with responsible licensee personnel, the inspectors concluded that the test anomalies were being properly addresses and that no concerns existed with a potential fire spreading beyond established boundaries. Resolution of the CO_2 discharge test anomaly remains an inspector followup item pending completion of the analyses and retest by the licensee and subsequent review of the results by the NRC. **(IFI 50-374/98-09-01)**

E8.2 (Closed) Violation 50-354/97-07-06: Failure to Include Five Struthers-Dunn Relays in the 10 CFR 50.49 Program

During a September 1997 inspection, the NRC identified a violation of the 10 CFR 50.49 requirement in that five normally energized safety-related Struthers-Dunn relays located in harsh environment had not been included in the list of equipment requiring environmental qualification (EQ) and, therefore, had not been replaced at the 5.04-years interval specified for the same relay in the same cabinet. The relays were 11 years old and three of them showed signs of age-related degradation, i.e., had visible debris from the magnetic vinyl plastic used as bearing pad material at the bottom of the relay case. Loss of the bearing pad material results in contact chattering and eventually in relay failure. The licensee had replaced the four of the relays and justified the qualified life of the fifth relay based on its not being continuously energized.

In their January 12, 1998, response to the NOV, PSE&G attributed the violation to personnel error and stated the relays had been added to Hope Creek EQ program and that recurring tasks had been developed for future relay replacement. The inspectors confirmed that the relays had been replaced, that recurring tasks for future relay replacements had been developed, and that the replacement intervals were correct and based on the new (silicon) bearing pad. This item is closed.

In their response to the NOV, PSE&G also noted that the qualified life (12 years) that had been established for the relays was appropriate and that the relays had been replaced prior to its expiration. The bases for the previous NRC conclusions regarding the qualified life of the relays were included in Inspection Report (IR) 50-354/97-07 and were discussed further with the licensee during a subsequent inspection of Struthers-Dunn relays in mild environment (IR 50-354/98-80). Also, because the licensee replaced the relays with other relays having silicon bearing pad material, the qualified life was no longer an issue. Nonetheless, for clarity purposes, those bases are repeated below:

Using the results of a Wyle qualification test, the Arrhenius equation for time-temperature dependency, and field measurements of relay coil and pad material temperature, PSE&G established (calculation STRDUN-ARRH-001, dated April 30, 1997) that the qualified life of normally-energized relays was 15.4 years. Then, for conservatism, they assigned to the relays a qualified life of 12 years. Although the calculation was technically acceptable, its results did not support the Hope Creek-specific life expectancy in that three of the four normally-energized relays in question were "degraded" after only 11 years, indicating that one or both the equation variables, i.e., activation energy and/or operating temperature, were incorrect. Therefore, PSE&G had no bases to conclude that the relays, in the conditions found and much less after 12 years, would have been able to perform their post-accident functions. As stated previously, because the relays were replaced with new relays having silicon bearing pad material, the service life of relays with magnetic vinyl bearing pads is no longer an issue.

E8.3 (Closed) Violation 50-354/97-07-07: Failure to Perform Adequate Relay Service Life Calculation.

During a September 1997 inspection, the NRC determined that the licensee had used an incorrect (lower) coil temperature rise in the design calculations to extend the service life of normally-energized, safety-related Agastat and Telemechanique relays in mild environment. The use of the lower temperature rise resulted in a longer relay service life. Engineering reviews of the design calculations had failed to identify the deficiency.

In their January 12, 1998, response to the NOV, PSE&G attributed the violation to personnel error and inattention to detail. To address this issue, the licensee discussed the deficiency with the personnel involved and recalculated the service lives of the relays involved. As a result of their evaluation, the licensee replaced some of the relays and developed recurring tasks for the replacement of others at the end of the calculated service life. The licensee's evaluation included a review of the relays safety functions. The inspectors reviewed the licensee's evaluation and the bases for the actions taken to address the violation and found them acceptable. They also reviewed the revised service life calculation and verified that the developed recurring tasks reflected the revised relay service life. This item is closed.

E8.4 (Closed) Unresolved Item 50-354/97-07-09: Circuit Breaker Failure Analysis

On September 10, 1997, during plant restoration, following a plant trip, a secondary condensate pump circuit breaker failed to trip on demand. Although the circuit breaker involved was not safety-related, because its manufacturer was the same as that of the equivalent safety-related breakers, the inspectors were concerned that a maintenance-related failure might also impact the safety-related breakers.

The inspectors' review of the licensee's root cause analysis, during the current inspection, determined that the breaker failure to open on demand was the result of inadequate preventive maintenance (PM). The licensee also determined that a contributing factor was the infrequent operation of the breaker that limited the lubrication of the trip roller carrier bearings. Because the breaker had not been overhauled and was infrequently operated, the grease had become hard and the trip mechanism bound up. The trip coil, unable to trip the breaker, had overheated and failed.

The applicable procedure required cleaning and PM every 36 months, but did not include disassembly of the lower mechanism. Therefore, the licensee only cleaned and lubricated surfaces that were reachable during the performance of the PM procedure. The breaker in question had undergone cleaning and maintenance in 1994, but disassembly of the operating mechanism had not been performed since original factory assembly. The PM procedure for the safety-related medium voltage breakers similarly did not specifically require disassembly.

In 1996, the licensee initiated a refurbishment program to be performed by the breaker manufacturer. At the time of the root cause analysis, 20 of the 44 safety-related and 22 of the 52 nonsafety-related breakers had been refurbished. In addition, they initiated a revision of the PM procedures to address the weaknesses.

At the time of the inspection, 38 breakers had never been refurbished. All of the breakers had date codes from 1979 to 1981 and, therefore, all were beyond the ten-year refurbishment cycle recommended by the manufacturer, as the licensee had recently established. Of these, 20 were safety-related and all were exercised at least once in 90 days. The inspectors discussed the current refurbishment schedule with the system engineer and determined that 10 of the 20 safety-related breakers would be refurbished before the next refueling outage, scheduled to begin on February 13, 1999. The other ten breakers were scheduled for refurbishment during the system outage window, before September 1999.

The inspectors also reviewed and discussed with the engineer the safety functions of the breakers and the results of selected previously refurbished breakers. They concluded that the refurbishment delay was not a major safety concern. In reaching their conclusion, the inspectors took into consideration that current experience with refurbished breakers showed only minor changes in the breaker close and trip voltage and speed and that there was no indication that the unrefurbished safety-related breakers were degraded and would fail to operate during an event. The lack of refurbishment of the nonsafety-related breakers was not a concern because they were associated with the nonsafety-related buses and would not perform an active safety function during an event that included a loss of offsite power.

The inspectors concluded that the licensee had properly addressed the breaker failure event; the root cause analysis had accurately evaluated the breaker failure modes; and the corrective actions were appropriate and properly managed. Because the failure involved a nonsafety-related breaker and because the result of ongoing refurbishment did not indicate pervasive degradation concerns potentially also affecting the safety-related breakers, the inspectors further concluded that no violation of NRC requirements had occurred regarding the breaker maintenance or in conjunction with the event. This item is closed.

E8.5 <u>(Closed) Violation 50-354/98-80-01</u>: Inadequate Test Control

On March 19, 1998, the NRC identified three examples where testing performed by PSE&G was not in conformance with the requirements of 10 CFR 50, Appendix B. Addressed below are the licensee's response to the Notice of Violation, dated August 6, 1998, and the actions taken by them to address the individual findings.

<u>Failure to Correct Capacity Test Results for Test Temperature.</u>

In November 1991, the licensee interrupted the performance capacity test for battery 1DD447 because of a loss of the load bank. The team found that the licensee had failed to account for this interruption and the consequent apparent capacity increase. Furthermore, the licensee failed to correct the capacity value for the actual test temperature, as required by the test procedure, to assure that the test requirements had been satisfied.

The licensee attributed the cause of the violation to inadequate procedural guidance regarding test interruptions that resulted in the use of the wrong electrolyte temperature. To address the NRC finding, the licensee reevaluated battery capacity performance test and confirmed that it met the Technical Specification requirements. In addition, they initiated action to revise the test procedure to address test interruptions. The inspectors reviewed the licensee's evaluation and found it acceptable. They also confirmed that the procedure revision had been entered in the corrective action program. The procedure revision was scheduled to be completed by October 30, 1998. This item is closed.

<u>Failure to Calculate Battery Average Capacity and Evaluate Need to Increase Test Frequency</u>.

In December 1995, test results indicated that the battery capacity had dropped 21.6% from the previous test. IEEE Standard 450-1995, to which the Hope Creek UFSAR states compliance, specifies that a decrease in battery capacity of more than 10% from the average of the previous tests should result in the licensee increasing the test frequency to 18 months. During the March 1998 inspection, the team found that, despite the large drop in capacity, the licensee had failed to calculate the average battery capacity, evaluate the test results, and determine whether the test frequency should be increased.

The licensee determined that the violation was the result of insufficient guidance being provided in the capacity test procedure. To address the technical issue, the licensee evaluated the previous test results, calculated the previous average battery capacity, and recalculated the capacity drop. They determined that the drop was less than 10%. Therefore, they concluded that an increase in test frequency was not necessary. To address the cause of the violation, they initiated a corrective action item to revise the test procedure and provide the necessary guidance. The inspectors reviewed the licensee's evaluation and found it acceptable. They also verified that the corrective action program included provisions for revising the test procedure to address this violation. They determined, as above, that the procedure revision was scheduled to be completed by October 30, 1998. This item is closed.

Failure to test the Control Room and Control Building HVAC System.

During a March 1998 inspection, the review of the licensee's practices for testing and inspecting the control equipment room supply (CERS) and the control area battery exhaust (CABE) systems determined that inservice testing of these systems was limited to checking the functionality of individual components under the preventive maintenance program and that the automatic standby features of these system were not being periodically tested as specified in Table 9.4-6 of the UFSAR.

The licensee, in their August 6, 1998, letter, attributed their failure to conduct the above test to over-reliance in the TS surveillance requirements also to satisfy the UFSAR testing commitments. To address the NRC finding, they evaluated the current tests and identified the need for establishing maintenance tasks to document pressure and flow capacities of the systems in question and to verify their autostart capability. Their evaluation also concluded that the systems operation and functionality testing demonstrated the acceptability of their performance. They established the development of the maintenance tasks for September 30, 1998. In addition, they planned to develop, by July 31, 1999, a testing program for other risk significant systems.

The inspectors reviewed the licensee's evaluation of the issue and found it acceptable. They also confirmed that the proposed activities to resolve the issue were included in the corrective action program. This item is closed.

E8.6 (Closed) Violation 50-354/98-80-04: Inadequate Design Control

On March 19, 1998, the NRC identified six examples where PSE&G's design control activities were not in conformance with the requirements of 10 CFR 50, Appendix B. Addressed below are the licensee's response to the Notice of Violation, dated August 6, 1998, and the actions taken by them to address the individual findings.

Use of Incorrect Temperature in the Selection of the Thermal Overload (TOL) Heaters for DC Loads.

In April 1991, the design basis for the thermal overload devices associated with the RCIC and HPCI loads powered from the dc motor control center was not correctly translated into design specifications, in that the licensee used as a design input for the reactor building ambient temperature 104°F rather than 148°F, as specified in the calculations of record.

In their response to the violation, PSE&G stated that the failure to use post-accident temperatures in the selection of thermal overload protection devices was the result of personnel error during the design of Hope Creek and incorrect assumptions on the bypass status of the devices during post-accident conditions. To address this finding, the licensee initiated action to revise the applicable calculation, E7.9, and reflect the correct temperature. In addition, they implemented a design change to remove the thermal overload protection for one HPCI and all RCIC motor-operated valves.

The inspectors reviewed the licensee's evaluation and determined that the maximum calculated switchgear room temperature was 117°F, rather than the originally specified 148°F. At the lower temperature, the selected thermal overload devices were acceptable. Nonetheless, calculation E-7.9 required revision to correct the design discrepancy and reflect the correct temperature. The licensee had scheduled the revision of the calculation for April 15, 1999, considering their expected NRC approval of a TS change request, as described below. The inspectors also reviewed design change package 4EC-3638 related to the bypassing of the RCIC and HPCI valve overloads. Based on the above review, a review of calculation E-7.9(Q), dated April 2, 1992, and verification that the calculation revision was in the corrective action program, the inspectors considered the resolution of the issue acceptable. This item is closed.

Use of Incorrect input Voltage and Efficiency in the Selection of Overload Protection Devices for the Battery Chargers.

The March 1998 inspection team found that, in April 1991, the licensee failed to consider the inverters minimum voltage and efficiency in the setting of the protective devices for the 20 kVA safety-related inverters. The licensee attributed the violation to a less than adequate engineering review of the design basis material and inadequate documentation in calculation E-7.9 of the bases for the setting selection.

Following the inspection, the licensee reevaluated the setting of the inverter protective devices and concluded that they were acceptable and consistent with the design basis documentation contained in calculation E-4.1. Therefore, no changes to the setting were needed. The licensee also concluded that calculation E-7.9, developed during the original plant design, required revision to correct the inverter loading discrepancies. The inspectors reviewed the licensee's evaluation as well as applicable sections of battery sizing calculation E-4.1(Q), Revision 11, and calculation E-7.9 (Q), Revision 0, and concluded that the licensee's evaluation and resolution of the issue was acceptable. As stated above, revision of calculation E-7.9 is planned for April 15, 1999. This item is closed.

<u>Relocation of Temperature Detectors without a Safety Evaluation of its Impact on the Technical Specification Setpoints.</u>

On March 12, 1996, the licensee approved a design change to move the temperature sensors in the supply air ducts of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) rooms downstream of in-duct heaters. However, they failed to verify the adequacy of the revised design and its impact on the TS-specified setpoints. Table 3.3.2-2 of the Hope Creek TS requires isolation of the RCIC and HPCI turbine steam supply, if the difference between the room exhaust and supply air temperatures exceeds 70°F. The allowable differential temperature specified by the table is 80°F.

The licensee, in their response to the violation, stated that, although an assessment of the changes impact on the TS-specified setpoint had been made during the design phase of the modification, responsible personnel had only implicitly discussed and not explicitly justified the licensing and design bases of the modification. The licensee re-reviewed the modification and the 10 CFR 50.59 applicability review and confirmed that the temperature sensor relocation was adequately justified. To avoid further violations the licensee had also planned to enhance the procedures controlling design change evaluations and to use the violation deficiencies in the training of technical support personnel.

The inspectors' review of performance improvement request (PIR) 980318152 determined that the associated evaluation had primarily addressed the adequacy of the 50.59 applicability review. The licensee had, nonetheless, evaluated the impact of moving the sensor downstream of the heater on the TS-specified differential temperature limits and concluded that it would have been minimal. The inspectors also reviewed the licensee's bases as well as applicable portions of calculations 11-85 (Q), Revision 1, and 11-22(Q), Revision 2, which provided the bases for the RCIC and HPCI isolation and concluded that the licensee's evaluation and resolution of the issue was reasonable. This item is closed.

<u>Failure to Assure that the UFSAR-Specified Battery Capacity Margin was Maintained.</u>

On August 27, 1997, the licensee issued an UFSAR change notice reducing the minimum battery design margin stated in Section 8.3.2.1.2.2 of the UFSAR from 5% to 0%. In making the change, they failed to verify the adequacy of the battery design under "less than optimum operating conditions," as stated in the UFSAR and specified in IEEE Standard 450-1975.

In their letter to the NRC, the licensee attributed the violation to personnel error and to a less than adequate review of the battery design basis information in their preparation of the 10 CFR 50.59 safety evaluation. To address the technical aspects of this finding the licensee confirmed that the battery capacities were currently acceptable and initiated action to issue a License Change Request (LCR) to increase the minimum battery room temperature licensing bases. This increase would permit them to recover the margin lost due to increased loading and conformance with the TS-specified minimum operating temperature. Regarding the inadequate review of the design basis information, the licensee stated that they had instituted a Safety Evaluation Independent Review Team (SEIRT), as they had previously done at Salem, to provide independent oversight of the 10 CFR 50.59 safety evaluation review process.

The inspectors reviewed the LCR and determined that licensee had requested to raise the battery room minimum design temperature from 60°F to 72°F. This was based on the fact that safety-related in-duct heaters maintained the battery room temperature at 77°F ±3°F. The increase in temperature increases the battery capacity margin by approximately 11%. Based on the above review, the inspectors concluded that the licensee's technical resolution of the battery margin was appropriate and acceptable. The inspectors also confirmed that steps had been taken for the review of safety evaluations developed for Hope Creek design changes by the SEIRT. This item is closed.

Failure to Maintain Chilled Water Temperature within Limits Specified in the HVAC Design Calculation.

The NRC found that the design bases for the heating and ventilation (HVAC) system of the control room and the safety-related panel room was correctly translated in surveillance procedures, in that the chilled water outlet temperature for these areas was set to be maintained between 43 and 47°F and between 45 and 49°F, respectively. The chilled water temperature limits specified in the UFSAR and in the HVAC design calculations were 45 and 47°F, respectively.

In their letter to the NRC, the licensee stated that the violation was due to engineering failure to include instrument accuracy into the design basis calculations. They evaluated the operation of the chilled water system against its design basis and confirmed that the current set points were acceptable. Therefore, they initiated action to revise the system design basis information. In addition, they evaluated several other systems to confirm that instrument accuracy had been properly addressed. The inspectors reviewed the evaluation performed by the licensee and calculation H-1-GM-MEE-1294, Revision 0, dated May 20, 1998, and confirmed that current setting of the chilled water systems were acceptable. The inspectors also confirmed that efforts were ongoing to evaluate setpoints to ensure that instrument accuracies were properly considered. This item is closed.

Failure to Evaluate the Impact on Cooler Performance Due to 18" Pipe Installed in Close Proximity of Inlet Flow Area.

In March 1996, the licensee approved a design change to install a RHR cross-over pipe in the "D" RHR room. However, prior to its installation, they failed to evaluate the revised design and ensure that the placement of an 18-inch RHR pipe in close proximity of the ECCS room cooler air inlet did not block and reduce the cooler air flow rates and impact the room cooler performance.

In their letter to the NRC, the licensee attributed the violation to personnel error and a less than adequate review and assessment of the design change. To address the technical issue, they evaluated the current installation and confirmed that it did not adversely impact the operability of the ECCS room cooler. Regarding the root cause of the violation they initiated action to revise the design change procedures to require documentation of the design and licensing basis information assessed during the review process. They also planned to address the violation during scheduled training of the engineering support personnel.

The inspectors reviewed the licensee's evaluation and confirmed that the impact of the pipe on the performance of the ECCS room cooler was minimal. The inspectors also determined that the licensee had issued a corrective action item against the calculation to address the additional resistance provided by the pipe to the flow as well as another discrepancy identified by the licensee during the root cause review of this issue. Based on the above review, the inspector concluded that acceptable actions had been taken to resolve the finding. This item is closed.

E8.7 (Closed) Violation 50-354/98-80-05: Inadequate Safety Evaluation

During the March 1998 inspection, the team identified two temperature and humidity recorders in the main control room and in the remote shut-down panel room that were not permanently installed. No written safety evaluation had been prepared which provided the bases for their conclusion that the instruments did not present a missile hazard to safety-related structures, systems and components in the rooms and that the facility change did not involve an unreviewed safety question.

In their response to the NOV, on August 6, 1998, the licensee stated that the violation was the result of personnel failure to follow procedures regarding control of measuring and test equipment, and lack of questioning attitude by station personnel. To address the seismic hazard concerns they removed the recorders from the affected areas and planned to evaluate the installation of other nonpermanent equipment in the control room. They, also, planned to evaluate potential enhancements of the applicable procedures and to address the lessons learned with operation, maintenance, and engineering support personnel.

The inspectors reviewed the results of the licensee's evaluation and subsequent corrective actions and found them acceptable. They also confirmed the removal of instruments from the affected areas. This was done in response to the NRC concerns as well as to the licensee's conclusion that the instruments were not needed on a permanent basis. The inspectors' review of the status of the proposed actions determined that a walkdown of the control room had been performed and potential seismic interactions had been identified and corrected. In addition, the results of the walkdown as well as the lesson learned from the violation had been rolled out to affected personnel. This item is closed.

E8.8 (Closed) Violation 50-354/98-80-08: Failure to Followup Identified Deficiencies of Emergency Lighting Units.

On March 23, 1997, fire protection technicians identified five inoperable emergency lighting units (ELUs) due to dead batteries. Although the batteries had been replaced, this had not occurred until after Quality Assessment (QA) identified, in August 1997, Fire Protection programmatic weaknesses that eventually resulted in the identification of approximately 43% of inoperable ELUs. The team's review of this issue identified several concerns, including the fire protection technicians' failure to notify the Nuclear or Senior Nuclear Shift Supervisor, as required by the fire protection procedure.

In their response to the Notice of Violation, dated August 6, 1998, PSE&G stated that they were not able to confirm that the operating staff had not been properly notified, but agreed that appropriate actions were not taken to address the ELU deficiencies on March 23, 1997. They attributed the cause to less than adequate administrative controls for impairment notification and less than adequate program oversight by Fire Protection Management.

The licensee's evaluation of the event identifies several corrective actions, including: (1) revision of procedure ND.FP-AP.ZZ-0010(Q), Fire Protection Impairment Program, to require notification documentation; (2) modification of the emergency lighting preventative maintenance program and issuance of a new procedure, HC.FP-PM.QB-0039(F), to address the PSE&G identified programmatic weaknesses; and (3) incorporation of the fire protection surveillances in the station surveillance scheduling process.

The inspectors reviewed the licensee's investigation and concluded that the evaluation of the program deficiencies had been thorough and that the resulting actions action appropriate. The inspectors also verified that the procedure and program changes had been acceptably completed. This item is closed.

E8.9 (Closed) Violation 50-354/98-80-09: Failure to Maintain Administrative Controls of Watertight Doors.

Contrary to the requirements of Hope Creek Technical Specification Section 3.7.3, on March 9, 1998, with the Delaware River water level still above the 95-feet PSE&G datum, operators reopened and failed to administratively control access through four service water intake structure watertight flood doors.

In their letter to the NRC, dated August 6, 1998, the licensee attributed the violation to a less than adequate understanding of the administrative control implementation requirements for flood protection during elevated river water condition. To correct the deficiency and avoid further violations, they revised procedure HC.OP-AB.ZZ-0139 to require the stationing of trained personnel at flood protection doors, if the doors need to be re-opened and the river water level conditions have the potential for affecting the operation of safety-related plant equipment. In addition, the licensee reviewed the violation and the procedure change with operating personnel.

The inspectors reviewed the revised procedure as well as the licensee's evaluation of the event and found them acceptable. The inspectors also determined that the information had been disseminated to the operating crew through a night order dated July 24, 1998. This item is closed.

E8.10 Review of Related UFSAR Sections

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the team reviewed the applicable portions of the UFSAR that related to the area inspected. The UFSAR wording was consistent with the observed plant practices and operating procedures.

V. Management Meetings

X1 Exit Meeting Summary

The teams presented the inspection results to members of licensee management at the conclusion of the inspection on September 8, 1998. The licensee acknowledged the findings presented during the meeting without comments.

The team asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Public Service Electric and Gas Company

R. DeNight	Supervisor, Specialty Engineering/Analysis
P. Duke	Hope Creek Licensing Engineer
V. Fregonese	Manager, Engineering
D. Garchow	Director. Design Engineering
S. F. Kobylarz	Supervisor, QA Engineering
C. Losnedahl	Hope Creek Reactor Engineer
M. Mannion	Manager, Nuclear Fuel
J. E. Moaba	Manager, Specialty Engineering
D. Nelson	Supervisor, Maintenance Engineering
D. Notigan	Supervisor, Hope Creek Fuels
D. R. Powell	Director, Licensing Regulation and Fuel
J. Priest	Hope Creek Licensing Engineer
B. C. Simpson	Senior Vice President, Nuclear Engineering
G. M. Stith	Hope Creek System Engineer

Nuclear Regulatory Commission

J. D. Orr	Resident Inspector
S. M. Pindale	Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-354/98-09-01	IFI	Resolution of CO_2 discharge test failure

Closed

50-354/96-09-03	VIO	De-Facto Modification Without Safety Evaluation. (Section E8.1)
50-354/97-07-06	VIO	Failure to Include Five Struthers-Dunn Relays in the 10 CFR 50.49 Program. (Section E8.2)
50-354/97-07-07	VIO	Failure to Perform Adequate Relay Service Life Calculation. (Section E8.3)
50-354/97-07-09	URI	Circuit Breaker Failure Analysis. (Section E8.4)
50-354/98-80-01	VIO	Inadequate Test Control (3 Examples). (Section E8.5)
50-354/98-80-04	VIO	Inadequate Design Control (6 Examples). (Section E8.6)
50-354/98-80-05	VIO	Inadequate Safety Evaluation (Section E8.7)
50-354/98-80-08	VIO	Failure to followup identified deficiencies of emergency lighting units. (Section E8.8)
50-354/98-80-09	VIO	Failure to maintain administrative controls of watertight doors. (Section E8.9)

LIST OF ACRONYMS USED

A/E	Architect/Engineer
ANS	American National Standard
AOV	Air-Operated Isolation Valve
APRM	Average Power Range Monitor
AR	Applicability Review/Action Request
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
BRDR	Battery Room Duct Reheat
CABE	Control Area Battery Exhaust
CACS	Containment Atmosphere Control System
CAG	Corrective Action Group
CARB	Corrective Action Review Board
CERS	Control Equipment Room Supply
cfm	Cubic Feet per Minute
CFR	Code of Federal Regulations
CM	Corrective Maintenance
CR	Condition Resolution
CRCW	Control Room Chilled Water
CROD	Condition Resolution Operability Determination
DABE	Diesel Area Battery Exhaust
DABR	Diesel Area Battery Room
DAPRS	Diesel Area Panel Room Supply
DCP	Design Change Package
DITS	Design, Installation, and Testing Specification
EACS	Equipment Area Cooling System
ECA	Engineering Change Authorization
ECP	Employee Concern Program
EDG	Emergency Diesel Generator
ELU	Emergency Lighting Unit
EPA	Electrical Protection Assemblies
FCD	Flow Control Diagram
UFSAR	Final Safety Analysis Report
GL	Generic Letter
HC	Hope Creek
HVAC	Heating, Ventilation, and Air Conditioning
HOAS	Hydrogen/Oxygen Analyzer System
HPCI	High Pressure Coolant Injection
IA	Instrument Air
IEEE	Institute of Electrical and Electronic Engineers
IFI	Inspection Followup Item
INPO	Institute of Nuclear Power Operations
IST	Inservice Test
kVA	kilo Volt-Amperes
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
MOV	Motor Operated Valve
M&TE	Measuring and Test Equipment
NCV	Non-Cited Violation

NRB	Nuclear Review Board
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NUREG	Report issued by the NRC
OSR	Offsite Safety Review
PCIG	Primary Containment Instrument Gas
PDR	Public Document Room
PRCW	Panel room chilled water
PRNMS	Power Range Neutron Monitoring System
PSE&G	Public Service Electric and Gas
psig	Pounds per Square Inch Gage
QA	Quality Assurance/Quality Assessment
RBTVB	Reactor Building to Torus Vacuum Breaker
RCIC	Reactor Core Isolation Cooling
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RWCU	Reactor Water Cleanup
SACS	Safety Auxiliaries Cooling System
SBO	Station Blackout
scfm	Standard Cubic Feet per Minute
SE	Safety Evaluation
SEIRT	Safety Evaluation Independent Review Team
SORC	Site Operation Review Committee
SRC	Switchgear room cooling
SRV	Safety Relief Valve
SSW	Station Service Water
STACS	Safety and turbine auxiliaries cooling system
TOL	Thermal Overload
TS	Technical Specification
TSHL	Temperature Switch High-Low
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UHS	Ultimate Heat Sink
VIO	Violation