UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-325/90-41 and 50-324/90-41

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket Nos.: 50-325 and 50-324 License Nos. DPR-71 and DPR-62

Facility Name: Brunswick 1 and 2

Inspection Conducted: October 2 - November 4, 1990

Lead Inspector: R. L. Prevatte 11/16/90
Date Signed

Other Inspectors: W. Levis
D. J. Nelson

Approved By: for R. E. Carroll, Acting Section Chief 11/19/90
Date Signed
Reactor Projects Branch 1
Division of Reactor Projects

SUMMARY

Scope:

This routine safety inspection by the resident inspectors involved the areas of maintenance observation, surveillance observation, operational safety verification, initial response to onsite events, onsite review committee, onsite followup of licensee event reports, review of 10 CFR Part 21 items, and action on previous inspection findings.

Results:

In the areas inspected, no programmatic weaknesses or significant safety matters were identified.

A non-cited violation for the failure to place a channel of the reactor protection system scram discharge volume water level high trip system in the trip condition after exceeding the Technical Specification two hour time limit for having this equipment in test was identified, paragraph 7.a.

Two minor deficiencies involving the lack of attention to detail by the control room operators were noted during routine plant tours, paragraph 4.

A review of the unit trip and degraded voltage event that occurred on September 27, 1990, identified four items where the inspector did not have sufficient information and/or the necessary resources to fully evaluate. These items will be referred to NRR for further review, paragraph 7.c.

Unit 1 was in a refueling outage during the reporting period. Unit 2 experienced an automatic trip on October 12 as the result of a blown fuse in the feedwater level control system, paragraph 5. The unit was restarted on October 18, 1990. The licensee appeared very conservative in their approach to unit restart and reduced power twice during equipment malfunctions that may have placed the unit at risk.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*K. Altman, Manager - Regulatory Compliance
F. Blackmon, Manager - Radwaste/Fire Protection
S. Callis, On-Site Licensing Engineer
T. Cantebury, Manager - Unit 1 Mechanical Maintenance
*G. Cheatham, Manager - Environmental & Radiation Control
M. Ciemnicki, Security
R. Creech, Manager - Unit 2 I&C Maintenance
J. Cribb, Manager - Quality Control (QC)
*W. Dorman, Manager - Quality Assurance (QA)/(QC)
*M. Foss, Supervisor - Regulatory Compliance
V. Grouse, Employee Relations
J. Harness, General Manager - Brunswick Steam Electric Plant
W. Hatcher, Supervisor - Security
R. Helme, Manager - Technical Support
J. Holder, Manager - Outage Management & Modifications (OM&M)
*M. Jones, Manager - On-Site Nuclear Safety - BSEP
R. Kitchen, Manager - Unit 2 Mechanical Maintenance
*B. Leonard, Manager - Training
*J. Leviner, Manager - Engineering Projects
J. McKee, Manager - QA
*J. Moyer, Technical Assistant to Plant General Manager
*P. Musser, Manager - Maintenance Staff
B. Poteat, Administrative Assistant to Plant General Manager
R. Poulk, Manager - License Training
*J. Simon, Manager - Operations Unit 1
W. Simpson, Manager - Site Planning and Control
S. Smith, Manager - Unit 1 I&C Maintenance
R. Starkey, Vice President - Brunswick Nuclear Project
R. Tart, Manager - Operations Unit 2
J. Titrington, Manager - Operations Staff
*R. Warden, Manager - Maintenance
B. Wilson, Manager - Nuclear Systems Engineering

Other licensee employees contacted included construction craftsmen, engineers, technicians, operators, office personnel, and security force members.

*Attended the exit interview

Acronyms and initialisms used in the report are listed in the last paragraph.

2. Maintenance Observation (62703)

The inspectors observed maintenance activities, interviewed personnel, and reviewed records to verify that work was conducted in accordance with approved procedures, Technical Specifications, and applicable industry codes and standards. The inspectors also verified that: redundant components were operable; administrative controls were followed; tagouts were adequate; personnel were qualified; correct replacement parts were used; radiological controls were proper; fire protection was adequate; quality control hold points were adequate and observed; adequate post-maintenance testing was performed; and independent verification requirements were implemented. The inspectors independently verified that selected equipment was properly returned to service.

Outstanding work requests were reviewed to ensure that the licensee gave priority to safety-related maintenance. The inspectors observed/reviewed portions of the following maintenance activities:

90-ALUC1	2C RBCCW Pump Rebuild
90-ASYH1	MSL Rad Monitor B Power Supply Changeout
90-PME411	Route on 2-CAC-1262
90-PZI395	Sample Pump Replacement for 2-CAC-4409
90-WFI414	Rosemount ATTU Output Voltage Check

Violations and deviations were not identified.

3. Surveillance Observation (61726)

The inspectors observed surveillance testing required by Technical Specifications. Through observation, interviews, and record review, the inspectors verified that: tests conformed to Technical Specification requirements; administrative controls were followed; personnel were qualified; instrumentation was calibrated; and data was accurate and complete. The inspectors independently verified selected test results and proper return to service of equipment.

The inspectors witnessed/reviewed portions of the following test activities:

1-MST-IRM12W	IRM Channels B, D, F, H Functional Test
1-MST-SRM22R	SRM Channels A and C Channel Calibration
PT-7.1.1.a	Core Spray Injection Check Valve Operability Test
PT-20.7.2	1-E11-F050B LLRT

Violations and deviations were not identified.

4. Operational Safety Verification (71707)

The inspectors verified that Unit 1 and Unit 2 were operated in compliance with Technical Specifications and other regulatory requirements by direct observations of activities, facility tours, discussions with personnel, reviewing of records and independent verification of safety system status.

The inspectors verified that control room manning requirements of 10 CFR 50.54 and the Technical Specifications were met. Control operator, shift supervisor, clearance, STA, daily and standing instructions, and jumper/bypass logs were reviewed to obtain information concerning operating trends and out of service safety systems to ensure that there were no conflicts with Technical Specification Limiting Conditions for Operations. Direct observations of control room panels and instrumentation and recorder traces important to safety were conducted to verify operability and that operating parameters were within Technical Specification limits. The inspectors observed shift turnovers to verify that system status continuity was maintained. The inspectors verified the status of selected control room annunciators.

Operability of a selected Engineered Safety Feature division was verified weekly by ensuring that: each accessible valve in the flow path was in its correct position; each power supply and breaker was closed for components that must activate upon initiation signal; the RHR subsystem cross-tie valve for each unit was closed with the power removed from the valve operator; there was no leakage of major components; there was proper lubrication and cooling water available; and conditions did not exist which could prevent fulfillment of the system's functional requirements. Instrumentation essential to system actuation or performance was verified operable by observing on-scale indication and proper instrument valve lineup, if accessible.

The inspectors verified that the licensee's health physics policies/procedures were followed. This included observation of HP practices and a review of area surveys, radiation work permits, postings, and instrument calibration.

The inspectors verified by general observations that: the security organization was properly manned and security personnel were capable of performing their assigned functions; persons and packages were checked prior to entry into the PA; vehicles were properly authorized, searched and escorted within the PA; persons within the PA displayed photo identification badges; personnel in vital areas were authorized; effective compensatory measures were employed when required; and security's response to threats or alarms was adequate.

The inspectors also observed plant housekeeping controls, verified position of certain containment isolation valves, checked clearances, and verified the operability of onsite and offsite emergency power sources.

An inspector found two minor control room deficiencies on Unit 2 not discovered by operators. On October 19, 1990, at approximately 7:30 a.m., the control switch key was found inserted in the control switch for Core Spray Valve F001A, Suppression Pool Suction Valve. Important component control switches on the control board are equipped with locks that require keys for operation to prevent inadvertent manipulation. Dummy keys are normally inserted with the color coded real keys attached. The valve was in its correct position, but had been manipulated for the performance of a surveillance test on the previous shift. Following the test, the key was not switched back. Prior to the inspector's discovery, three control operators and one trainee had performed their board walkdown without detecting the error. The inspector informed the operators of the problem and the key was promptly switched.

On October 24, 1990, the inspector observed that one of six APRM GAFs indicated 0.00 on the 7:00 a.m. process computer P-1 printout. Normal GAF values are 0.98 to 1.00. The printout also flagged APRM B as a failed sensor. The process computer calculates the GAF from a ratio of calculated power and APRM indicated power. The 0.00 value observed by the inspector indicates that a meaningful GAF could not be calculated due to a problem with calculated or indicated power. The inspector reviewed previous hourly P-1s, which also indicated GAFs of 0.00 back to 3:00 a.m., which had a 4.17 value. When questioned, the on-duty operators were unaware of and could not explain the discrepancy. Eventually the operators determined that a surveillance test prior to 3:00 a.m. for APRM Channel B caused the computer to cease its "scan" of APRM B due to inconsistent values calculated when the APRM output was manipulated during the test. The APRM was placed back in service following the surveillance test, but the computer "scan" was not reset. This did not affect the operability of the APRM, and no other indications in the control room were affected.

GAF values are the most consistent indication of APRM reliability. Although the APRM trend chart recorders would also indicate an actual APRM problem, the recorders cannot trend all six APRM channels simultaneously. Therefore, it is important for operators to monitor the GAFs.

These two examples, while not safety significant, indicate that continued diligence by operators is needed in monitoring control room indications.

Violations and deviations were not identified.

5. Initial Response to Onsite Events (93702)

Unit 2 Scram

Unit 2 was operating at 100 percent power on October 12, 1990, when a fuse blew in the FWLCS resulting in a loss of power to a number of components in the control circuit. The loss of power gave the appearance of low reactor water level which caused an increased demand signal to the RFPs and a trip signal to the reactor recirculation pump runback circuits. The

"A" RFP also locked up due to the loss of power resulting in the "B" RFP responding to the master controller demand for increased feed flow. Twenty-six seconds after the fuse blew an actual high water level condition was reached which caused a turbine trip and reactor scram.

The lowest reactor water level reached was approximately 117 inches. Group 2, partial Group 3 and Group 6 isolations were received. Reactor recirculation pumps tripped due to the low level condition. RCIC auto started and injected and HPCI auto started but did not inject due to the short duration of the level transient. HPCI was subsequently used to raise water level to the normal band.

During post trip recovery, the operators had difficulty in placing a RFP in service. Because of the loss of power to the FWLC circuit, the master controller was still demanding 100 percent output. When 2B RFP was placed back in service, in automatic, the pump increased speed to 5700 RPM and discharge pressure increased to 1700 psig. With the flowpath isolated due to feed pump trip recovery actions, the 4 and 5 feedwater heater relief valves lifted and began releasing steam into the feedwater heater room. When steam was reported as being released in these rooms, the feedwater heater inlet valves were shut securing the leak. The FWLC system was also placed in single element and selected to channel B, which then restored level feedback to the controller.

The fuse that blew in the FWLC circuit was a Gould Shawmut A25Z3 fuse and was similar to the fuse which blew in this circuit on August 16, 1990, that also resulted in a reactor scram. The licensee has subsequently replaced the Gould Shawmut fuses in the FWLC circuitry with Bussman MIN fuses which are the type installed in Unit 1. The Gould Shawmut fuses were installed as an Appendix R modification to provide separation between safe shutdown circuits and other associated circuits of concern. Subsequent review by the licensee documented in EER 90-0262, October 14, 1990, determined that separation of this circuit was not required.

The inspector will review the licensee's corrective actions taken with respect to the feed pump operation and the fuse failure when the LER is issued.

6. Onsite Review Committee (40700)

The inspectors attended selected Plant Nuclear Safety Committee meetings conducted during the period. The inspectors verified that the meetings were conducted in accordance with Technical Specification requirements regarding quorum membership, review process, frequency, and personnel qualifications. Meeting minutes were reviewed to confirm that decisions/recommendations were reflected in the minutes and followup of corrective actions was completed.

Violations and deviations were not identified.

7. Onsite Followup of Licensee Event Reports (92700)

The below listed LERs were reviewed to verify that the information provided met NRC reporting requirements. The verification included adequacy of event description and corrective action taken or planned, existence of potential generic problems, and the relative safety significance of the event. Onsite inspections were performed and concluded that necessary corrective actions have been taken in accordance with existing requirements, license conditions, and commitments, unless otherwise stated.

a. (Closed) LER 1-90-16, Operation Prohibited by Plant Technical Specifications During SDV Maintenance and Surveillance Activities. On September 17, 1990, the Unit 1 SF authorized the performance of 1MST-RPS27R, RPS Scram Discharge Volume High Water Level Channel Functional Test and Channel Calibration. In conjunction with the test, the SF also authorized work to be performed in accordance with the instructions of WR/JO 90-AMCT1, which replaced the electronic printed circuit board for level switch 1-C11-LSH-4516C. The SF entered a tracking LCO for this work believing that only one input to channel A2 would be disabled.

Subsequent review of this work on September 20, 1990, by a different shift foreman, revealed that the MST disables the trip function of channel A2 during a portion of the test. A two hour time limit is allowed by Technical Specification 3.3.1 for a channel to be disabled during surveillance testing provided that the other channel in the same trip system is operable. After the two hour period, the channel must be returned to operable status or placed in the tripped condition. Because of the maintenance actions performed on 1-C11-LSH-4516C, the two hour time limit was exceeded during performance of the MST on September 17, 1990. The SF did not place the channel in the tripped condition, as required by Technical Specification 3.3.1, because he did not realize that the A2 channel was disabled during the testing. The channel was inoperable for 2 hours and 20 minutes.

In their investigation, the licensee noted several contributing conditions. First, it is not appropriate to perform corrective maintenance while performing surveillance tests. The two hour time period allowed by Technical Specifications for inoperable channels applies to surveillance testing and not corrective maintenance. A Standing Instruction was put in place until permanent procedure revisions are made that prohibit corrective maintenance to be performed during surveillance testing under the two hour time constraint. In addition, the MSTs will be revised to enhance the specific instruments that will be made inoperable by jumper, and training for the appropriate people will be performed.

The failure to place channel A2 of the RPS SDV water level high trip scram signal in the tripped condition is a violation of TS 3.3.1: Failure to Place Channel A2 In the Tripped Condition, (325/90-41-01).

This licensee identified violation is not being cited because criteria specified in Section V.G.1 of the NRC Enforcement Policy were satisfied.

b. (Closed) LER 1-90-017, Unit 1 High Pressure Scram During Performance of Turbine Control/Stop Valve Tightness Test. A unit 1 scram, from high pressure, during the performance of Turbine Control/Stop Valve Tightness Test, occurred on September 27, 1990, and was discussed in inspection report 90-37. At the time of that report, the licensee was still investigating the event and preparing a LER. The investigation was subsequently completed and LER 90-017 was issued on October 26, 1990. The LER provided a description of the events surrounding this occurrence and contained recommendations to prevent recurrence. This event occurred with the unit at approximately 22 percent power while in the process of shutting down for a refueling outage. The licensee had started the planned Periodic Test (PT) 40-2-10, Turbine Control/Stop Valves (TCV/TSV) Leak Tightness Testing. The event occurred due to erroneous procedural guidance provided by the vendor, General Electric, in GEK-25406A and defective switches on the TSVs, which allowed the TCVs to open when the TSVs were closing. The turbine BPV's open demand signal was limited by the maximum combined flow circuitry of the turbine control system. The closure of the TSVs without the BPVs being able to open caused reactor pressure to increase to the SCRAM setpoint. A detailed licensee review of this event has determined that this procedure contained weaknesses which are applicable to both nuclear and fossil plants which have used this procedure/guideline to develop their plant tests. The licensee had made this information available to other utilities through "Network" and have indicated that GE may issue an information letter on this item.

c. (Closed) LER 2-90-15, Unit 2 Reactor Scram Due to Loss of Excitation on Main Generator. A scram occurred on Unit 2 on September 27, as a result of the loss of excitation on the main generator. This item was also discussed in inspection report 90-37, and the licensee was investigating this event at the close of the previous inspection period. The licensee subsequently completed their investigation into the event and provided the details in LER 90-015, dated October 26, 1990. The inspectors have reviewed the LER and other licensee documentation associated with the event and discussed this matter with NRR. Based on this review and discussion with the licensee and NRR, the inspectors are unable to determine the following: (1) was the capacity and stability of the offsite power system prior to and immediately after the Unit 2 trip adequate to provide acceptable voltages to handle a design basis event in Unit 2 and a safe shutdown on Unit 1; (2) does the licensee's loss of voltage protection system (relays), which only sheds the safety-related equipment off the "E" buses, provide adequate protection for system voltages less than the setpoint of the degraded voltage relays; (3) if the offsite power voltage drops to a level that is inadequate to start safety-related loads, will the 10.5 second degraded voltage relay time delay result in a delaying safety bus transfer to the diesel generator such that

the plant may be outside the boundary of this safety analysis; and (4) does CP&L provide adequate direction and guidance to system load dispatchers to ensure that adequate voltage is maintained at all nuclear units? These items will be referred to NRR for review. Pending the outcome of the above, these items will be tracked as an inspector followup item: Adequacy of Offsite Power, (325,324/90-41-02).

One non-cited violation was identified.

8. 10 CFR Part 21 Items (36100)

(OPEN) 325,324/P2188-01 - Worn Shaft Gear Failures in Size 2 Limitorque Actuators and Also in Fisher Supplied H3BC Actuators. The inspector discussed this item with the licensee. A review of licensee records could not determine that this report had ever been received. They contacted Limitorque who verified that the information had been provided, so it was apparently lost. The licensee has entered this item into FACTS and has assigned responsibility for investigation and resolution. This item will be reviewed further as information becomes available.

(CLOSED) 325,324/P2188-04 - Reinstalling Foxboro Controller Circuit Cards May Cause 100 Percent Output and Subsequent Transient to Occur. This controller has been identified in BWR Recirculation Flow Control System DCS-88080301. This Part 21 was sent by the Foxboro Company to the Perry Nuclear Power Plant. It related to SPEC 200, Model 2AC-D+44 controller card with its associated 2AX+RM removable manual card. This notice was not sent to Brunswick since they do not use these units in their recirculation flow control system. A review of parts and installation by the licensee has verified that these units are not installed or used in spare parts at Brunswick.

(CLOSED) 325,324/P2189-01 - Brown Boveri K-Line, K-225 Through K-2000 Circuit Breakers Delivered Prior to 1974 Had Rebound Spring Added to Slow Close In. This item identified that testing had discovered that the above breakers may fail to function properly in that persistent sine dwell vibration could occasionally cause the slow close bar to move into a position such that the breaker, when called on to close, could slow close rather than closing normally. Adding a rebound spring to the slow close lever will prevent the slow close bar from vibrating to the undesired position. Brunswick was identified as a plant that had received shipment of these breakers. A survey by the licensee found that 10 electrically operated breakers of this type were installed in the plant. Substations 3L and 4L located in the hot machine shop have K-1600 incoming main breakers. These units were manufactured in 1972 and do not contain rebound springs but are used in non-safety applications to provide power to the warehouse, maintenance shops and office buildings. Emergency substations E5, E6, E7, and E8 were found to have K-3000 breakers and the Part 21 does not apply to the breakers. The four remaining breakers are installed as crosstie breakers on substations E5, E6, E7, and E8. These are K-1600 breakers manufactured before 1972 and do not contain rebound

springs. However, these breakers are administratively controlled in the racked out position and verified racked out as part of normal system surveillance testing. The breakers may be closed at the discretion of the shift supervisors when both units are in mode 4 or 5 and other TS requirements are met. No credit is taken in the accident analysis for use of these tie breakers. Based upon the above, the licensee discovered that the use of these components at Brunswick does not pose a substantial safety hazard and, therefore, is not reportable under 10 CFR Part 21. Since this defect would only occur under seismic event conditions and the components are not used for safety applications, the licensee does not have any current plans to install recommended springs in the substations used for the warehouse, maintenance shop, and office buildings. The licensee plans to purchase and install the springs in the crosstie breakers for substations E5, E6, E7, and E8 at the next scheduled maintenance period after receipt of the springs.

(CLOSED) 325,324/P2189-05 - PT-21/Germane to Safety from GE: Susceptibility of Weld Between Core Spray Line and Thermal Sleeve to IGSCC. GE recommends that welds be included in IE Bulletin 80-13 surveillance. This item was evaluated by GE and the licensee and determined to be germane to safety, but not reportable. The licensee has implemented the vendor recommendations and included this item in their surveillance program to be tested each refueling outage under Periodic Test, Core Spray/Feedwater Visual Examination, PT-90.1, and reported under IEB 80-13. The inspector verified that these reports had been submitted as required.

(CLOSED) 325,324/P2189-06 - PT-21/Germane to Safety from GE: Concerns with Core Neutron Flux Monitoring and Reactor Protection During Refueling. This item was the result of NRC questioning the conservatism of center-spiral reloading because the SRMs were not on scale and, therefore, not monitoring neutron flux changes during a refueling at the Brown's Ferry Plant. GE performed an evaluation of this event and concluded that this event did not constitute a substantial safety hazard and was not reportable under the context of 10 CFR Part 21. However, they did conclude that this issue was germane to safety. Based on the above, RICSIL No. 039 was issued by GE on February 10, 1989, to alert BWR owners that interim recommendations would be provided. These interim recommendations were: (1) that during refueling, the neutron monitoring system should be capable of continuously monitoring changes in neutron flux in the region of the core where fuel is being loaded or control rods are being removed to provide operators with indications of an approach to criticality; (2) that the RPS should be capable at all times of reliably initiating a reactor scram based on inputs from the neutron flux detectors; and (3) that the recommendations of SIL 372 and SIL 68 should be followed when refueling interlocks are bypassed. The inspector reviewed the licensee's Engineering Evaluation Report (EER) 89-0022, dated January 16, 1989, that evaluated this event and the vendor recommendations, the licensee's Refueling Fuel Handling Procedure FH-11, Volume IX, Revision 41, and Engineering Procedure Guidelines for Preparation of Core Component Sequence Sheets, ENP-24.12,

Volume XX, Revision 8, to verify that the vendor recommendations had been implemented. This review indicates that Procedure FH-11 provides Administrative Control and direction over core reloads, that all control rods will be fully inserted during fuel movement, that the source range monitors will be operable and providing on scale indications, that refueling interlocks will be operable and not bypassed or jumpered out, and that the core loading will progress in a sequence which ensures that the SRMs have accurate indication of changes in neutron levels. These procedures, therefore, implement the vendor recommendations and appear to be satisfactory. EPRI is currently conducting a study on this issue. It is anticipated that this study will be completed in late 1990. The licensee has indicated that they will review and implement the recommendations of that study as appropriate to the Brunswick Plant.

(CLOSED) 325,324/P2189-18 - SMB Actuators Found to Have Melamine Torque Switches That Undergo Post Mold Shrinkage and Cause Cam Binding. Melamine Torque Switches Found to be Not Qualified. The licensee reviewed this item and determined it to be applicable to BSEP. A decision was made to replace the applicable torque switches on PCIS valves during the 1989-90 refueling outage for Unit 2 and on Unit 1 during its refueling outage in 1990-91. The remaining torque switches were scheduled for replacement when routine maintenance is performed on the remaining valve actuators with all replacement work completed by July 3, 1992. The inspector reviewed the completed work request for PCIS replacement accomplished on Unit 2 during the past outage, and the work scheduled for Unit 1 during the current outage. The listing of work scheduled under the routine maintenance program for the remaining torque switches was also reviewed. It appears that the licensee has determined which torque switches require replacement and have the program needed to complete these activities underway with an established completion date.

(CLOSED) 325,324/P2189-12 - PRE-1981 SMB-000 and PRE-1976 SMB-00 Cam Type Torque Switches Can Fail as a Result of Stationary Contact Screws Loosening on Side of Torque Switches That Had Fiber Spacers. Two failure modes were reported. The first type failure resulted when one of the screws in the contact bridge came loose resulting in premature tripping of the torque switch. The second failure resulted when both stationary contact screws loosened and the contact bridge raised with the contact fingers maintaining continuity on the torque switch. The licensee's evaluation concluded that no events of this nature have occurred at BSEP and, due to the low number of failures reported, concluded that the safety significance was low. A licensee's review of safety-related SMB-00 and SMB-000 valves at BSEP determined that these switches, which were made of melamine or phenolic materials, had been or were in the process of being replaced with new fiberite torque switches as a result of 10 CFR 21.89-18. The remaining SMB-000 and SMB-00 torque switches that were made of fiberite, were verified as having been replaced since 1985; with the exception of one switch, 1-E21-F001A-MO, that is planned to be worked under Work Request 89-AZIA1. The new torque switches and any SMB-000 purchased since 1980 and SMB-00 switches purchased since 1976, do not include fiber spacers and do not have this problem. Based on the above,

it appears that the licensee has identified all equipment affected by this item and has either completed or established documentation to inspect and replace the remaining items.

(CLOSED) 325,324/P-2190-04 - Rosemount Resistance Bridges Can Exhibit Premature Long Term Degradation Under Certain Combinations of Humidity, Power and Temperature. Two of the units referenced in the above, serial numbers 0067897 and 0067898, were provided to BSEP. Neither of these units have been installed. They have been placed under administrative hold in stock under CP&L Part Number 731-758-12. They will be returned to the vendor for replacement when the new units on order are received. This item is closed.

Violations and deviations were not identified.

9. Action on Previous Inspection Findings (92701) (92702)

(CLOSED) Violation 325,324/88-18-05, HPCI/RCIC High Steam Line Flow Instruments Inoperable. The inspector reviewed the licensee's response to the Notice of Violation and Civil Penalty dated January 27, 1989. The specific corrective actions taken to resolve the instrument's setpoints was detailed in LER 1-88-14, which was inspected and closed in Report No. 90-37. The violation was issued because of the inadequate corrective actions taken to resolve the issue when it was first identified. In their response, the licensee committed to revising their corrective action program along with establishing a program to effectively implement the BSEP system engineering concept.

Weaknesses in the licensee's corrective action and system engineering programs were also noted in the Diagnostic Evaluation Team report. As a result, the licensee included these specific areas into their IAP program which resulted in further NRC inspections to followup on the program's implementation. The licensee's new corrective action program was inspected in Report No. 90-31. The report stated that the actions committed to in the IAP were completed. The effectiveness of the new program was not assessed and will be evaluated in future inspections.

System engineering program improvements were inspected in Report No. 90-16. The report stated that the appropriate procedures and programs were in place to correct deficiencies in this area. Based on the inspections completed to date, which address the required corrective actions of the violations and the additional inspections planned to assess the effectiveness of the licensee's corrective action, this item is closed.

(CLOSED) Violation 325,324/88-24-03, Silicon Bronze Bolts Corrective Actions. The inspector reviewed the licensee's response to the Notice of Violation and Civil Penalty dated January 27, 1989. This violation, along with 325,324/88-18-05, were included in EA-149 and constituted the 2 examples of inadequate corrective action. The specific actions taken with regard to the silicon bronze bolt issue is detailed in LER 1-88-06, which

was inspected and closed out in Report No. 90-37. As discussed in the closeout of Violation 88-15-05, the licensee's actions to resolve corrective action program deficiencies have been and are continuing to be inspected as part of the followup to the licensee's IAP. Based on these inspections, this item is closed.

(CLOSED) Violation 325,324/88-34-01, Inadequate Design Control Related to RCIC Steam Exhaust Check Valve. This violation concerned the Unit 1 and 2 RCIC Steam Exhaust Check Valve, E51-F040. Plant Modifications 81-274 and 81-275, replaced the Unit 1 and 2 valves which included discs with design pressures of 25 psig. Based on the calculated peak containment pressure during a DBA of 49 psig, and containment design pressure of 62 psig, the discs were under rated. The inspector reviewed the licensee's response to the violation and supporting documentation. EER-88-0461 was written to evaluate the design pressure discrepancy and concluded that the discs were sufficient. This was based on successful local leak rate testing at 49 psid and documentation from the valve manufacturer stating that the discs would "withstand a pressure of 62 psig at 248 degrees F for a sustained period of time". The EER concluded that these values are greater than the containment system requirements for the DBA and higher than credible turbine exhaust operating pressures. The cause of the design error was that the containment accident pressure and LLRT pressure were not evaluated; only operational exhaust pressures were considered during the design process. Revisions to applicable administrative and engineering procedures have added formalized checklists and training requirements for safety reviewers.

(CLOSED) Violation 325,324/88-38-01, Failure to Control Combustibles in Restricted Plant Areas. The inspector reviewed the licensee's response to the Notice of Violation and supporting documentation. The licensee stated that the reason for the violation was that personnel not familiar with fire-retardant wood requirements obtained wood from outside the protected area for use as forms to install a plant modification. A contributing factor was that the modification instructions did not specify the type of material to be used for the forms. The licensee enhanced the existing controls to allow only fireproof material and/or fire retardant wood to be used within the plant protected area. An improvement in identification markings for fire retardant wood to ensure this type of wood, including cut up pieces, is readily identifiable for use in the protected area. The training lesson plan for the plant construction support group annual training was revised to address this topic.

(CLOSED) Unresolved Item 325,324/88-05-01, Service Water System Operating Mode Concerns. Other inspections of this issue are documented in Inspection Reports 89-09, 89-12 and 89-14. As a result of these inspections and others performed on the service water system, a Notice of Violation and Imposition of Civil Penalty was issued on January 26, 1990, for failure to take adequate corrective actions for identified service water deficiencies. Example A of the violation describes the inadequate evaluation performed when determining system operability with the single failure concerns

associated with the SW-V106 valve. Further inspection of this issue will be performed in the closeout of Violation 325,324/89-34-47.

(CLOSED) Unresolved Item 325/88-34-02, Reactor Vessel Water Level Wide Range Indication Anomolies. This item was also discussed in inspection report 325,324/90-02. Currently the licensee is replacing Rosemount transmitters in Unit 1 and is delaying any further action on the wide range indicators pending completion of the replacements. The licensee expects some indication difference with the new transmitters. The inspector determined that no regulatory issues existed. Therefore, this item is closed.

(CLOSED) Unresolved Item 325,324/88-38-02, Failure to Include All LPCI and Suppression Pool Cooling Flow Path Boundary Valves in Surveillance Program. This item was also discussed in inspection report 89-05 and was expanded to include core spray system valves. This item concerned the PTs that meet the monthly TS surveillance requirements of ECCS systems for verification that each valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position. The licensee originally disagreed with the inspector on what was a "flow path" valve. Specifically, minimum flow valves, vent isolation valves, valves normally out of position in standby lineup (i.e., E11-F027A(B), RHR Suppression Pool Spray Isolation), or flow path boundary valves were not included in the surveillances. The licensee has subsequently revised the PTs to include these valves. No occurrences are known where these valves have been found out of position due to their omission in the PTs. This item is closed.

(CLOSED) Unresolved Item 325,324/90-17-02, Potential Inoperability of CBEAF System. Based on inspector's questions regarding the differential pressure measurement technique used to determine the CBEAF system operability, the licensee changed the test procedure to more accurately measure the differential pressure (see inspection report 90-02). Subsequent tests verified that previous test results may have been inaccurate in determining that a positive pressure existed in the control building relative to the outside atmosphere. Based on the licensee's previous analysis and documentation from 1985, the erroneous differential pressure measurement technique has minimum safety significance, therefore, this item is closed.

(CLOSED) IFI 324/88-15-05, Normal Position for SW-V117, Nuclear Header to Vital Header Isolation Valve. As a result of extensive review and analysis of the service water system design, the normal position of the SW-V117 was changed from closed to open. The valve position was changed to allow a RHR room cooler to be placed in service affording the service water pumps minimum flow protection under worst case single failure scenarios. The Service Water System Operating Procedures, 1-OP-43, Revision 31, and 2-OP-43, Revision 68, require that the valve be open. The licensee plans to keep the valves open until the service water pump thrust bearing modifications are completed. This work is currently scheduled to begin in 1992.

(CLOSED) IFI 325,324/88-38-05, Licensee Activities Related to Correcting Keepfill System Discrepancies. The licensee has determined that the primary cause of keepfill system problems are bent stems on pressure control valves due to over-pressurization downstream. Several causes of over-pressurization have been identified. Currently three of twelve keepfill systems are inoperable. Modifications are scheduled for the current Unit 1 outage and the next Unit 2 outage to replace pressure control valves. This item is closed based on the licensee's identification of the causes of the keepfill system discrepancies and planned corrective actions.

Violations and deviations were not identified.

10. Exit Interview (30703)

The inspection scope and findings were summarized on November 5, 1990, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Description/Reference Paragraph
325/90-41-01	NON-CITED VIOLATION - Failure to Place Channel A2 In the Tripped Position (paragraph 7.a).
325,324/90-41-02	IFI - Adequacy of Offsite Power (paragraph 7.c).

11. Acronyms and Initialisms

AO	Auxiliary Operator
APRM	Average Power Range Monitor
ATTU	Analog Transmitter Trip Unit
BPV	Bypass Valve
BSEP	Brunswick Steam Electric Plant
BWR	Boiling Water Reactor
CBEAF	Control Building Emergency Air Filtration
CP&L	Carolina Power & Light Company
DBA	Design Basis Accident
ECCS	Emergency Core Cooling System
EER	Engineering Evaluation Report
ENP	Engineering Procedure
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
F	Degrees Fahrenheit
FACTS	Facility Automated Commitment Tracking System
FWLCS	Feedwater Level Control System
GAF	Gain Adjustment Factor
GE	General Electric
HP	Health Physics

HPCI	High Pressure Coolant Injection
IAP	Integrated Action Plan
I&C	Instrumentation and Control
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IFI	Inspector Followup Item
IGSCC	Intergranular Stress Corrosion Cracking
IPBS	Integrated Planning, Budgeting and Scheduling
IRM	Intermediate Range Monitor
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRT	Local Leak Rate Test
LPCI	Low Pressure Coolant Injection
MSL	Main Steamline
MST	Maintenance Surveillance Test
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PA	Protected Area
PCIS	Primary Containment Isolation System
PNSC	Plant Nuclear Safety Committee
PSID	Pounds per Square Inch Differential
PSIG	Pounds per Square Inch Gauge
PT	Periodic Test
QA	Quality Assurance
QC	Quality Control
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RFP	Reactor Feed Pump
RHR	Residual Heat Removal
RICSIL	Rapid Information Communication Service Information Letter
RPM	Revolutions Per Minute
RPS	Reactor Protection System
SDV	Scram Discharge Volume
SF	Shift Foreman
SRM	Source Range Monitor
STA	Shift Technical Advisor
TCV/TSV	Turbine Control Valve/Turbine Stop Valve
TS	Technical Specification
URI	Unresolved Item
WR/JO	Work Request/Job Order