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

Report Nos.: 50-325/85-16 and 50-324/85-16

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: June 1-30, 1985

Inspectors: AK Harden for 7/12/85
W. H. Ruland Date Signed

AK Harden for 7/12/85
L. W. Garner Date Signed

AK Harden for 7/12/85
T. E. Hicks Date Signed

Approved by: AK Harden for 7/12/85
P. E. Fredrickson, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine safety inspection involved 300 inspector-hours on site in the areas of followup on previous enforcement matters, maintenance observation, surveillance observation, operational safety verification, onsite review committee, onsite Licensee Event Report review, followup on inspector identified and unresolved items, plant modifications, and refueling activities.

Results: A violation was identified - inadequate surveillance test procedure for the refueling hoist slack cable interlock.

## REPORT DETAILS

1. Persons Contacted

P. Howe, Vice President - Brunswick Nuclear Project
C. Dietz, General Manager - Brunswick Nuclear Project
T. Wyllie, Manager - Engineering and Construction
G. Oliver, Manager - Site Planning and Control
J. Holder, Manager - Outages
E. Bishop, Assistant to General Manager
L. Jones, Director - QA/QC
M. Shealy, Acting Director - Training
M. Jones, Acting Director - Onsite Nuclear Safety - BSEP
J. Chase, Manager - Operations
J. O'Sullivan, Manager - Maintenance
G. Cheatham, Manager - Environmental and Radiation Control
K. Enzor, Director - Regulatory Compliance
B. Hinkley, Manager - Technical Support
L. Boyer, Director - Administrative Support
V. Wagoner, Director - IPBS/Long Range Planning
C. Blackmon, Superintendent - Operations
J. Wilcox, Principal Engineer - Operations
W. Hogle, Engineering Supervisor
W. Tucker, Engineering Supervisor
B. Wilson, Engineering Supervisor
R. Creech, I&C/Electrical Maintenance Supervisor (Unit 2)
J. Moyer, I&C/Electrical Maintenance Supervisor (Unit 1)
R. Kitchen, Mechanical Maintenance Supervisor (Unit 2)
R. Poulk, Senior NRC Regulatory Specialist
D. Novotny, Senior Regulatory Specialist
W. Dorman, QA - Supervisor
W. Hatcher, Security Supervisor
W. Murray, Senior Engineer - Nuclear Licensing Unit

2. Exit Interview

The inspection scope and findings were summarized on June 28, 1985, with the general manager. A violation described in paragraph 5 was discussed in detail. The licensee acknowledged the findings without exception. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Followup on Previous Enforcement Matters (92702)

(Open) Violation 325, 324/82-10-02: Failure to Implement Independent Verification per NUREG 0737 Item I.C.6. This item will remain open pending inspectors' review of the licensee's present implementation of I.C.6.

(Closed) Violation 325,324/84-35-01: Failure to Establish Adequate Implementing Procedure for Operating Procedure OP-32.

(Closed) Violation 324/82-39-01: Procedure Does Not Contain Isolation Valve for Low Condenser Vacuum Instrument. The inspector verified that the subject valves are included in the current revision (No. 32) to OP-30. As part of the operating procedure rewrite program, the licensee verified or included as necessary similar instrument isolation valves. This program was completed in December 1983.

(Closed) Violation 324/82-39-04: Failure to Bypass Low Condenser Vacuum Above 500 psig.

(Closed) Violation 324/82-39-03: Failure to Follow Procedure EI-13, Mode Switch Not Locked.

(Closed) Violation 325/83-30-01 and 324/83-30-01: Surveillance Test Method To Time Standby Gas Treatment System Inlet and Outlet Dampers Is Inadequate. The licensee's response, dated October 12, 1983, to the Notice of Violation committed to: (1) revise PT-15.7 to use MCC position indication or visual observation of valve travel, (2) revise OP-10 to provide a caution note that the control room position lights do not reflect actual valve position and such can be obtained from MCC and, (3) review to determine if similar conditions exist on other valves.

The inspector verified that the current revisions, revision 14 to PT-15.7, revision 5 to Unit 1 OP-10 and revision 24 to Unit 2 OP-10 contain changes as committed. The review disclosed no additional similar conditions on other valves.

(Closed) Violation 325/82-10-05 and 324/82-10-05: Failure to Implement Maintenance Procedure MI3-3A34. The licensee failed to perform corrective actions as described in their response (dated May 24, 1982) to the Notice of Violation. Another Notice of Violation was issued in report 325, 324/82-45. Corrective action will be inspected as part of the followup on the latter violation. This item is considered closed for administrative purposes.

No violations or deviations were identified.

4. Maintenance Observation (62703)

The inspectors observed maintenance activities 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; QC 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 and authorizations (WR&A) were reviewed to ensure that the licensee gave priority to safety-related maintenance.

The inspectors observed/reviewed portions of the following maintenance activities:

Work Request and Authorization (WR&A) No. 1-M-85-2109, 1C Residual Heat Removal (RHR) Service Water Booster Pump, Damaged Bearing.

WR&A No. 1-M-85-2199, 1C RHR Booster Pump High Vibration, High Bearing Temperature.

WR&A No. 2-M-85-2384, Core Spary Minimum Flow Bypass Valve Repairs, 2E21-F031A.

WR&A No. 0-E-85-1793, Scram Solenoid Valve Replacement Unit 1.

No violations or deviations were identified.

5. Surveillance Observation (61726)

The inspectors observed surveillance testing required by Technical Specifications. Through observation 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:

*1MST-RP622R, Main Steamline Isolation Valve Closure Channel Calibration.

PT-12.3.1, Emergency Diesel Generator Annual Inspection.

PT-12.4, Diesel Generator Alarms, Trips and Trip Bypass Test.

PT-85-071, Special Procedure for Response Time for Relays E41A-K4 and E41A-K5.

PT-12.2D, No. 4 Diesel Generator Monthly Load Test.

1MST-RHR21R, Residual Heat Removal, Low Pressure Coolant Injection, Core Spray System, High Pressure Coolant Injection (HPCI) High Drywell Pressure Trip Unit Channel Calibration.

PT-16.2-2, Primary Containment Volumetric Average Temperature, Rev. 1.

*Detailed Review

a. HPCI Response Time Testing

On June 21, 1985, the Maintenance Surveillance Test (MST) procedure writing groups discovered a discrepancy in the current High Pressure Coolant Injection (HPCI) System initiation response time test, Periodic Test (PT) 45.3.4. This PT failed to adequately calculate the response time for the HPCI System initiation logic as required by Technical Specification 4.3.3.3.

PT-45.3.4, HPCI initiation response time test was undergoing an upgrade process by the MST rewrite group (see LER 1-85-03). The procedure covered Technical Specification 4.3.3.3, which states that the Emergency Core Cooling System (ECCS) response time of each ECCS function shall be demonstrated to be within the required limits at least once per 18 months. The definition of ECCS response time is "that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function."

The current test method adds two partial response times to obtain the total time. One partial response time test is run for the sensor and associated logic response times. (There should be two trains tested, one for low low reactor water level and one for high drywell pressure.) The second test is a single HPCI autostart response time test. This section uses a simulated signal injection at an appropriate overlap point with the low low reactor water level sensor logic. However, there was not an adequate overlap point with the high drywell pressure logic train to consider the response time of all its associated relays. The response time of the E41-K4 and E41-K5 and E41-K5 relays (high drywell pressure initiation relays) was not considered in the total high drywell pressure response time test.

The licensee conducted a Plant Nuclear Safety Committee (PNSC) meeting concerning the implications of this problem. The PNSC concluded that: (1) The design response time of the individual relays in question as compared to the total response time required by Technical Specifications was minimal. The requirement is less than or equal to 30 seconds. The last HPCI system response time test was 29.64 seconds. The licensee had no history of this type relay failing a response time test. The design response time of these relays was .08 seconds. Based on this information, the HPCI system was not considered inoperable. (2) Plant conditions were: Unit 2 power level at approximately 90%, 'A' loop of Residual Heat Removal (RHR) System under Limiting Condition for Operations (LCO), No. 3 Diesel Generator under LCO for maintenance. To response test the relay, wires would be lifted and HPCI declared inoperable. This would have put the unit outside its LCO action statement and into LCO 3.0.3, which requires the reactor shutdown to hot standby in 6 hours. The 'A' RHR loop was expected back in service that afternoon, and No. 3 Diesel Generator returned Monday, June 24, 1985. (3) Plant management decided to test HPCI once a special

procedure (SP-85-071) was written and No. 3 Diesel Generator was returned to service.

Technical Specification 6.8.1.c requires that adequate procedures be implemented covering surveillance and test activities of safety related equipment. Technical Specification 4.3.3.3. requires an ECCS response time test to be conducted every 18 months. Contrary to this requirement, the licensee did not establish an adequate procedure for Technical Specification 4.3.3.3., in that the testing procedure, PT-45.3.4., HPCI Initiation Response Time Test, did not test all the relays in the high drywell pressure initiation logic. This violation of NRC requirements was identified by the licensee and meets the requirements of 10 CFR Part 20, Appendix C, Section V.A. A Notice of Violation will not be issued.

b. Refuel Hoist Slack Cable Interlock

On June 8, 1985, defueling of the Unit 1 core was halted following continued problems with the main hoist refuel grapple. The problems resulted from slack cable interlock actuations during normal refuel grapple operations. While investigating the interlock malfunction and conducting corrective maintenance, PT 18.1, Refueling Position Interlock Check, was found to be inadequate.

Technical Specification Surveillance Requirement 4.9.6.d states that each crane or hoist used for the movement of fuel be demonstrated operable by insuring operation of the slack cable cutoff when the load was less than 50 plus or minus 25 pounds for the mast fuel gripper. Technical Specification 6.8.1.a, requires the licensee to establish implementing procedures recommended in Appendix 'A' of Regulatory Guide 1.33, November 1972. Item H.2 of the guide specifies that procedures are required for each surveillance test, inspection and calibration listed in the Technical Specification.

Contrary to the above, PT-18.1 did not demonstrate that the slack cable cutoff occurred within the required tolerance. Step 7.3.5 of PT-18.1 only functionally checked the operation of the slack cable cutoff but did not verify the setpoint of the interlock. This inadequacy constitutes a violation 325,324/85-16-01; Inadequate Surveillance Test Procedure for the Refueling Hoist Slack Cable Interlock. The licensee has revised the procedure to include a setpoint verification and completed the test satisfactorily prior to resuming defuel operations.

This procedure deficiency was also identified in 1983 during a Technical Specification review by the Onsite Nuclear Safety Group, but an unsatisfactory resolution of the issue left the procedure as is. The failure to resolve the issue properly appears to have resulted from a lack of understanding of the Technical Specification and the fuel grapple operation. Onsite Nuclear Safety is currently reviewing additional comments generated during the 1983 review to identify similar problems.

c. Diesel Generator Rocker Arm Assembly Nuts

During observation of PT-12.3.1, for Emergency Diesel Generator No. 4, the inspector observed the some of the nuts which attach the rocker arm assembly to the cylinder head appeared to be cracked. Inspection by the licensee revealed 21 nuts which were suspect and were replaced. During a subsequent performance of PT-12.3.1, for Diesel Generator No. 3, the licensee replaced 30 similar nuts. The licensee has sent several nuts to the Harris Energy Center for metallurgical examination. Preliminary results indicate that, if the nut does not fail during installation, it will not fail during diesel generator operation. The inspector will review the final report when issued. The licensee plans to inspect diesel generators 1 and 2 during July and replace nuts as necessary.

6. Operational Safety Verification (71707)

The inspectors verified conformance with regulatory requirements by direct observations of activities, facility tours, discussions with personnel, review 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 room, shift supervisor, clearance 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 Specifications Limiting Conditions for Operations. Direct observations were conducted of control room panels, instrumentation and recorder traces important to safety to verify operability and that parameters were within Technical Specification limits. The inspectors observed shift turnovers to verify that continuity of system status was maintained. The inspectors verified the status of selected control room annunciators.

Operability of a selected ESF train was verified by insuring that: each accessible valve in the flow path was in its correct position; each power supply and breaker, including control room fuses, were aligned for components that must activate upon initiation signal; removal of power from those ESF motor-operated valves, so identified by Technical Specifications, was completed; there was no leakage of major components; there was a proper lubrication and cooling waster available; and a condition did not exist which might 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 procedures were followed. This included a review of area surveys, radiation work permits, posting, and instrument calibration.

The inspectors verified that: the security organization was properly manned and that security personnel were capable of performing their assigned functions; persons and packages were checked prior to entry into the protected area (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.

No violations or deviations were identified.

7. 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 actins was completed.

No violations or deviations were identified.

8. Onsite Review of Licensee Event Reports (92700)

The listed Licensee Event Report (LER) was 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 it was concluded that necessary corrective actions had been taken in accordance with existing requirements, licensee conditions and commitments. The following report is considered closed:

(Closed) LER 2-83-35; Reactor Water Cleanup differential flow indicator was indicating high because instrument had a low output signal.

No violations or deviations were identified.

9. Followup on Inspector Identified and Unresolved Items (92701)

(Open) IFI 325,324/84-35-02; Post-Trip Review. The inspectors reviewed OI-22, Plant Incident and Post-Trip Investigation, Rev. 10, to ensure that an adequate post-trip review program was established. OI-22 was also reviewed as part of the Generic Letter 83-28 inspection (85-14). No trips have occurred since the latest revision to OI-22. This item will remain open pending implementation review of OI-22.

(Closed) IFI 324/82-08-07 and 325/32-08-07, Post Potentially Contaminated Areas Until Survey Shows Otherwise. The event was reviewed with the appropriate personnel at the time of the event. The inspector verified that step 8.6.4 of the current revision (No. 4) to procedure E&RC, Posting of Areas/Materials, instructs personnel as follows: "...area that has the potential to be contaminated from occurrences (...leak from a...potentially contaminated system...) should be posted as a Contaminated Area until a survey...can be performed...". This procedure adequately addresses the inspector's concerns.

No violations or deviations were identified.

10. Design, Design Changes and Modifications (37700)

The inspectors reviewed selected modifications to verify that: activities were conducted in accordance with appropriate specifications and drawings; appropriate administrative controls were implemented; and acceptance testing was appropriate. The inspectors reviewed portions of the following modifications:

Plant Modification 82-219Q - "B" Loop RHR Service Water.

Plant Modification 82-287H,O - Reactor Instrument Penetration Valve Replacement.

MI-16-35A,C - HFA Relay Reconfiguration and Replacement.

Plant Modification 1-82-271, Level Transmitter LT-N026A&B Replacement.

While reviewing a proposed Plant Modification 84-058, the licensee discovered a wiring error for the RHR "2D" Pump Loss of Suction Trip in the remote shutdown mode. The specific error involved an HFA relay in the 2-E11-F009 valve logic.

This error would have prevented the operation of the 2D RHR pump in the shutdown cooling mode from the remote shutdown panel. No other modes of operation were affected. The problem was corrected immediately under a trouble ticket.

The licensee conducted additional system logic drawing review and field investigation of the remote shutdown systems. During this review, numerous drawing errors, logic errors and incorrectly wired HFA relays were discovered. Many of the drawing and logic errors had been corrected in the field but the as-built condition was not reflected on the prints. Except for the 2-E11-F009 valve logic error, no other problem identified to date affected the operability of any equipment in any mode of operation.

The problems appear to have been caused by an installation deficiency associated with the contact numbering convention for the HFA relays, depending on whether the relay was flush or recessed mounted. The plant modification installing the remote shutdown system in 1977 did not adequately verify that: (1) the wiring was in accordance with the drawings, (2) the as-built condition was correct or, (3) the acceptance testing was adequate.

Since this installation, the Plant Modification Procedure (ENP-03) has been revised and in the inspector's judgement would preclude this situation from occurring again.

The licensee's engineering staff has proposed the following corrective actions:

a. Review logic and drawings associated with the remote shutdown circuitry for wiring and drawing discrepancies.

b. Complete as-built verification of remote shutdown circuitry and identify discrepancies between the as-built and plant drawings.

c. Initiate and perform required repairs to correct errors through the use of plant modifications or work requests.

d. Perform acceptance testing through the use of special procedures that will test the remote shutdown system in all modes of operation.

e. Clarify drawing orientation of HFA relays to distinguish between the two relay configurations.

Prior to this event, the licensee had already undertaken a program to test the different modes of remote shutdown (1984 - 1985). To date, only the Reactor Core Isolation Cooling System had been satisfactorily tested. Plant operating conditions prevented further testing.

Inspector Followup Item (IFI) 325,324/85-16-02; Remote Shutdown Panel Wiring Deficiencies, will track the licensees' actions and progress in this area.

No violations or deviations were identified.

11. Refueling Activities (60710)

During the licensee's defueling operations, the inspectors verified that surveillance testing required by Technical Specifications was current and that the licensee's fuel handing procedures were implemented. The following items were verified:

a. Selected fuel bundle movements and storage locations.

b. Core monitoring during defuel operations was in accordance with Technical Specifications.

c. Vessel water level was maintained in accordance with Technical Specifications.

d. Reactor mode switch position was as required by Technical Specification.

e. Continuous communications were maintained between the refueling platform and the Control Room and the Control Room operators were cognizant of the applicable procedure steps.

f. Health-Physics personnel maintained constant coverage of all fuel moving activities, ensuring area dose rates, contamination levels and airborne samples were within required tolerances.

No violations or deviations were identified.