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

Report Nos. 50-325/85-22 and 50-324/85-22

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: July 1 - 31, 1985

Inspectors: HRKrug Jr 12 AUG 85
for W. H. Ruland Date Signed

HRKrug Jr 12 AUG 85
G. L. Paulk Date Signed

for HRKrug Jr 12 AUG 85
L. W. Garner Date Signed

for HRKrug Jr 12 AUG 85
T. E. Hicks Date Signed

for

Approved By: HRKrug Jr 12 AUG 85
for P. E. Fredrickson, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine safety inspection involved 237 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 Reports review, plant modifications, and inadvertent core spray initiation.

Results: Two violations were identified: Inadequate Corrective Action Relating To Annunciator Procedures and Hydraulic Control Units (HCU's) Not Installed As Per Drawing; two unresolved items, Drywell Temperature Calculation and Unusual Event Due to Fuel Pool Overflow.

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SUMMARY DETAILS

1. Persons Contacted

   Licensee Employees

   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 & 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, Principle 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

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

   NRC Resident Inspector

   G. Paulk, Senior Resident Inspector, Brown's Ferry

2. Exit Interview (30703)

   The inspection scope and findings were summarized on August 7, 1985, with the general manager. Two violations, described in paragraph six, were 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 the inspection.

3. Followup on Previous Enforcement Matters (92702)

(CLOSED) Violation 325, 324/85-03-02; Inadequate Procedure, Operating Procedure (OP) 6.12, Condensate Phase Separator Operating Procedure.

The inspector verified that: (1) OPs 6.12, 6.13, 6.14, 6.15 and 6.16 were changed to include cautions to direct operating personnel to close loading dock isolation valves from the radwaste control room upon indication of unusual circumstances or conditions; (2) PT-45.1, Loading Dock Transfer Lines Leak Test, was revised to incorporate necessary precautions to preclude line freeing; (3) Radwaste operating personnel were trained on the procedural changes and this event. Also, the radwaste loading dock was illuminated and the flex hose drained when not in use.

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:

WR&A 1-M-85-2431, SGTS (for PT-15.1.2)

MI-16-528, Fisher Butterfly Valves

PM-84-195, N2 Backup System

PM-84-299, Service Water Phase II Inspect/Repair/Replacement

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:

- 2MST-RHR21M, Rev. 0
- PT 16.2-2, Primary Containment Volumetric Average Temperature
- Daily Surveillance Requirements

a. Primary Containment Average Air Temperature Determination

The licensee has not yet found any record which supports the use of certain coefficients used to calculate primary containment average temperature. Technical Specification 4.6.1.6 requires the licensee to determine primary containment volumetric average air temperature at least every 24 hours. PT-16.2-2 implements T.S. 4.6.1.6. Thermocouple readings between certain elevations are averaged and multiplied by a volumetric coefficient for the specific containment volume monitored. Values from five volumes are added together to obtain the volumetric average temperature. The licensee has been unable to produce records to support the values used for the volumetric coefficients in PT-16.2-2. The licensee continues to interview current and former employees and search records for supporting information for the coefficients. Until the licensee can support the calculations and the coefficients used in PT-16.2-2, this item is Unresolved: Drywell Temperature Volumetric Coefficients, (325, 324/85-22-03).

b. Daily Surveillance Requirements Review

During a tour on July 1, 1985, the inspector noted that the 1600-2400 hours shift foreman failed to review and initial the Daily Surveillance Requirements (DSR) log for June 30, 1985. On July 17, 1985, the inspector noted that the steam jet air ejector radiation monitor readings (D12-RM-K601A and B) were not taken between 2200-2400 hours on July 16,1985. The reactor operator also failed to initial the page. These omissions were discussed with the shift operations supervisor and plant management, the identified omissions were corrected.

No violations or deviations were identified.

6. Operational Safety Verification (71707 and 71710)

The inspectors verified conformance with 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 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 proper lubrication and cooling water 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 policies/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 a clearance, and verified the operability of onsite and offsite emergency power sources.

a. Inadequate Annunciator Procedures

During a review of the Plant Operating Manual Annunciator Procedures (APs), a generic deficiency was identified. Contained in the action statements for many of the procedures are references to Emergency Instructions (EIs) which were designed to provide the operator with followup actions and recovery steps. However the EIs had been completely replaced over one year earlier (approximately May 1984) with either Emergency Operating Prucedures (EOP's) or Abnormal Operating Procedures (AOPs) depending on the emergency. The licensee's plan was to update the APs sometime thereafter. In the interim period, a cross reference was generated and provided to the operators in the control room. The cross reference would be used until the APs could be revised. However, this document had not been incorporated into an approved procedure and was inadvertently removed from the control room. If one of the annunciator procedures in question was then used, the operator would have to identify the referenced EI with a new procedure by reviewing the AOP, EOP procedure index and determining which new procedure was applicable. Since the titles of the new procedures did not necessarily match the old procedure titles, this type of evolution could be time consuming and allow for errors.

Consequently this program failed to be adequately implemented in that most of the procedures reviewed, which required changes, were not yet updated nor did the operators have a readily available means to cross-reference the old EI's with the new EOP's and AOP's.

Examples of this problem are as follows:

Annunciator Panel	Annunciator Description	Action Statement Reference
A-03; 1-7	RHR System II Actuator	EI-1.2 Rupture Inside Drywell, EI-08 Abnormal Reactor Water Level, EI-5.2 Loss of Primary Containment
A-03; 3-4	Reactor Core Isolation Cooling Pump Discharge Flow Low	EI-07 Reactor Core Isolation Cooling System Failure
A-05; 3-5	Reactor Vessel Pressure High	EI-4.1 MSIV Closure, EI-31 Reactor Scram
UA-03; 6-1	High Activity Process Off Gas Vent Pipe	EI-26 Off Gas High Radiation, EI-27.3 Abnormal Release of Radioactivity Airborne

UA-03; 6-6	Area Radiation Stack Filter House High	EI-23.4 High Radiation in Accessible Areas
UA-04; 1-4	Reactor Feed Pump Turbine Tripped	EI-09 Condensate and Feedwater System Failure
A-06; 4-6	Steam Tunnel High Temp System B	EI-31 Reactor Scram, EI-20 Main Steam Line Leaks
UA-15; 2-1	E1 Bus Undervoltage	EI-15.1 Station Blackout Operation, EI-15.2 Degraded Auxiliary Electrical Power Operation

10 CFR 50 Appendix B, Criterion XVI, as implemented by FSAR Section 17.2.16, Corrective Action, requires measures be established to assure that conditions adverse to quality (i.e., deficiencies) are promptly identified and corrected. Technical Specification 6.8.1.c requires that the licensee maintain and implement procedures specified in Reg. Guide 1.33, November 1972. Item "E" of the guide requires that procedures for correcting abnormal, offnormal or alarm condition be implemented.

Contrary to the above, a condition adverse to quality was not adequately corrected in that most annunciator procedures which required changes, caused by the introduction of Emergency Operating and Abnormal Operating Procedures and the elimination of the Emergency Instructions, were not updated and corrected or an alternate means adequately provided to clarify reference discrepancies contained in the procedures. This is a violation, inadequate corrective action relating to annunciator procedures (325, 324/85-22-01) of NRC requirements. The licensee has since provided the operators with a readily accessible cross-reference list while the annunciator procedures are changed.

Also, ANSI 18.7, 1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, Section 5.2.15, Review, Approval and Control of Procedures, requires that plant procedures be reviewed no less frequently than every two years to determine if changes are necessary or desirable. The licensee's Administrative Procedures, Volume I Book I, Section 5.6, implements this ANSI requirement and specifies that the reviewer "Attest that the procedure remains accurate, effective, and useful for its intended purpose; or otherwise, revisions are initiated to correct identified deficiencies." The licensee is committed to comply with ANSI 18.7, 1976.

The licensee did conduct a review (completed in April 1985), as required, but failed to make any procedure changes as a result of it. One reviewer did identify several deficiencies in three of the annunciator panels and submitted OI-28, Preparation and Review of Operating Procedures, changes. At the time of the inspection, none of the changes had been implemented.

b. Hydraulic Control Unit Problems

A routine inspection of the Control Rod Drive (CRD) Hydraulic Control Units for Unit 2, revealed the following items, which were not in accordance with plant design drawing G. E. 919D615:

1) HCU racks Nos. 18-51, 14-19, 02-27 and 06-31 had support to foundation bolts and/or nuts disengaged.

2) HCU rack No. 30-43 had 2 of the 4 rack support to foundation bolts missing.

3) All bolts, lockwashers and washers on the HCU rack supports did not appear to be cadmium plated as required.

4) HCU rack No. 18-35 support to foundation bolt had what appeared to be a welding ground pigtail attached under it.

In addition caps were missing from directional control valves on HCU's Nos. 18-07, 34-31, 18-31 and 42-11. The area also exhibited poor housekeeping in that miscellaneous materials were in, on or between the HCU racks. These items included: parts of valves, a lead brick, a stainless steel bar, rusty washers, nuts and bolts, rolls of duct tape and trash.

Items 1) and 2) constitute a violation of 10 CFR 50 Appendix B, Criterion V (325, 324/85-22-02), hydraulic control units not installed per drawings. Subsequent inspections by the licensee revealed similar discrepancies on Unit 1.

c. Diesel Generator Walkdown

During a detailed walkdown on Diesel Generator No. 1, prior to post-maintenance testing, the inspector found four valves which had no tags on them and were not contained in the Operating Procedure OP-39 valve check list. Two of these were valves associated with calibration of the pressure switch which provides for actuation of the shutdown cylinder to close the fuel racks during engine shutdown. These three valves are addressed in the appropriate maintenance procedures. A similar condition existed on the other three diesel generators. The remaining valve was installed in parallel with an air regulator valve in the starting circuit. The other diesel generators have a check valve in this application as shown in the technical manual. The licensee is evaluating this "as found" condition. The licensee has

indicated that OP-39 will be revised to include these valves as appropriate. This is an Inspector Followup Item: OP-39 Revision (325, 324/85-22-04).

d. Hurricane Bob

On July 24, 1985, the remnant of Hurricane Bob, downgraded to a tropical storm, past approximately 100 miles inland of the Brunswick site. Only periods of heavy rain were experienced at the site. Plant operations were unaffected. Unit 1 was in refuel and Unit 2 was operating near full power. The licensee had taken preparatory action in case the hurricane did approach the site. This included tracking of the hurricane and establishing criteria for activation of the technical support center and shutdown of Unit 2. Unit 2 would have begun shut down if hurricane force winds were expected at the site within four hours. No such action was required.

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 actions was completed.

No violations or deviations were identified.

8. Onsite Review of Licensee Event Reports (92700 and 90712)

The listed Licensee Event Reports (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, licensee conditions and commitments. The following reports are considered closed:

(CLOSED) LER 1-83-01; Pipe sides of weld B32 showed pin hole cracks attributed to IGSCC. Future action addressed in response to IEB 82-03. This defect was identified during 82-03 inspections. Bulletin 82-03 was closed for Unit 1 in IE inspection report 85-08.

(CLOSED) LER 1-83-04; Two crack indications in the heat affected zone on the pipe side of weld downstream of the loop discharge valve. These defects were identified during IEB 82-03 inspections. Bulletin 82-03 was closed for Unit 1 in IE inspection report 85-08.

(CLOSED) LER 1-83-49; Two crack indications exist in the heat affected zone on the pipe side of weld B32-RECIRC-28-A-14 and B32-RECIRC-28-B-8 due to IGSCC. These defects were identified during IEB 83-02 inspections.

Further licensee and NRC actions concerning IGSCC are addressed in Generic Letter 84-11.

(CLOSED) LER 1-85-024; Manually initiated isolation of Units 1 and 2 common control building heating ventilating air conditioning system.

(CLOSED) LER 2-83-38; An inoperative and rod block signal was received due to defective multiplexer cards.

(CLOSED) LER 2-83-40; Residual Heat Removal System flow transmitter had no output response to an applied test signal.

No violations or deviations were identified.

9. Design, Changes and Modifications (37700)

On July 1, 1985, a construction engineer working on Unit 1 suppression chamber modifications discovered the suppression chamber spray nozzles were not installed. The unit was in cold shutdown undergoing a refueling outage.

The suppression chamber spray header system is a subsystem of the containment spray system, and is an operational mode of Residual Heat Removal System (RHR). With the RHR system in containment cooling mode of operation, the RHR pumps are aligned to pump water from the suppression pool through the RHR system heat exchangers where cooling takes place by transferring heat to the unit service water system. Flow returns to the suppression pool via the full flow test line.

Under post-accident conditions, the containment cooling subsystem provides additional redundancy to the core standby cooling systems. Approximately 5 percent of the flow from the heat exchangers may be directed to the suppression chamber spray ring to cool any noncondensible gases collected in the free volume above the suppression pool. Analysis shows (refer FSAR Section 6.2.1.1.3.2.1), that containment spray is not necessary to provide adequate containment protection during the design basis accident. Therefore, containment spray failure was not a safety problem (refer FSAR Section 6.2.2.2).

The licensee determined that the suppression chamber spray nozzles have never been installed. A review of startup testing turnover packages revealed no apparent reason for the deficiency. New nozzles have been ordered and will be installed prior to suppression chamber closeout and plant startup. This corrective action will be verified during subsequent routine inspections.

No violations or deviations were identified.

10. Core Spray Initiation and Overflow of Spent Fuel Pool (93702)

On July 30, 1985 at 2236 hours, a spurious low vessel water level LOCA signal initiated available ECCS systems on Unit 1. This resulted in over-filling of the spent fuel pool and subsequent release of approximately 25,000 gallons of water into the Reactor Building. Safety systems which automatically actuated included three emergency diesel generators, core spray loop 1A and group I isolation. Other systems which would normally have started were under clearances for either maintenance or modification. Unit 1 was in an outage with the vessel defueled, fuel pool gates removed and the vessel cavity flooded up. Full power operations of Unit 2 were unaffected by the event. Description of the sequence of events, root cause and preliminary corrective actions, damage assessment and cleanup status follows:

6-15-85 Clearance No. 886 boundary changed to allow venting and hydrostatic testing of instrument 1-B21-LT-N026A as directed by plant modification 82-271. Clearance designated No. 886A.

7-01-85 Clearance tags for 1-B21-LT-N026A removed from No. 886A to allow testing on another modification.

7-30-85 Clearance tags for 1-B21-LT-N026A not rehung prior to resuming venting of instrument.

2236 Venting of 1-B21-LT-N026A causes slight depressurization of variable instrument leg containing 1-B21-LT-N031A and C. These latter instruments initiate a low level 2 group I isolation signal and a low level 3 LOCA signal. The control operator (CO) observes the group I isolation annunciator and auto starting of diesel generators 1, 3 and 4. The shift foreman thought radiography in the main streamline isolation valve pit may have commenced. A senior control operator (SCO) went to the backpanel to check radiation monitor status.

2237 SCO reports fire in backpanel. Fuel pool high level alarm is received. CO requests radwaste to maximize reject from fuel pool.

2238 CO begins control board walkdown. Group I is reset. Core spray logic system failure annunciator is in alarm. Health physicist on refuel floor reports fuel pool level is at ventilation ducts which surround the upper portion of the fuel pool walls.

2239 CO trips running CRD pump. Core spray pump 1A is found running and is manually tripped.

2240 Extinguishment of the backpanel fire commences.

2245 Fire is reported out in the core spray logic panel. Two HFA relays had smoked. No flaming had occurred. No other components were damaged by the fire.

7-31-85 At 0903 hours, the Licensee determined that they were in an unusual event based upon their emergency guideline procedure. Per the Shift Operating Supervisor judgement, it was determined that plant conditions exist that warrant increased awareness on the part of the plant operating staff. This was predicated on the fact that a large area of the reactor building had become contaminated and that if the water dried out, there was a potential for the areas to become airborne.

8-01-85 At 0830 the unusual event was terminated based upon the progress being made by the decontamination teams.

The root cause of the event was determined to be a problem with restarting a modification procedure after it had been interrupted. In addition, its effects were amplified by another problem. Installation of the nitrogen backup supply to the drywell modification had inadvertently installed AC type HFA relays into the core spray system logic which is powered by DC. Thus, when the spurious LOCA signal occurred, these relays burned up causing loss of power to the core spray logic. With the logic de-energized, all initiation logic lights and annunciators associated with the core spray initiation were inoperable. Hence, the operator was not immediately notified of the core spray injection. Another possible annunciator, Core Spray/RHR Pump Running, was already lit since the RHR pump was being used for shutdown cooling.

Interim measures taken by the licensee until long term corrective action can be developed, include: (1) double sign-on of clearances on modifications to assure proper tags are in place; (2) engineers will review test procedures to identify untested devices such as the HFA relays which may have already been placed in service and 3) reverification of clearances on tests which are interrupted. Final corrective actions will be included in the LER submittal.

Overflowing of the fuel pool into the ventilation ducts resulted in: damage to the duct work including approximately 150 feet of duct which fell onto the floor next to the standby gas treatment trains; water on the -17', 20' and 50' elevations of the reactor building; non-safety related MCC 1XJ requiring de-energization; partial loss of reactor building lighting; and malfunctioning of the E11-F017A valve (outboard low pressure coolant injection valve). When the ducting fell, it bent one safety related cable tray; however, no cables were damaged in the tray. The problem with the E11-F017A valve is still being determined. The Licensee has decontaminated the floors, inspected the inside of panels and MCC's and taken actions necessary to resume outage work. Outage work was restarted on August 5.

This event is considered an unresolved item pending further NRC review of the event and contributing circumstances (325/85-22-05).

No violations or deviations were identified. One unresolved item is noted.