UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report Nos. 50-324/81-14 and 50-325/81-14

Licensee: Carolina Power and Light Company
411 Fayetteville Street
Raleigh, NC 27602

Facility Name: Brunswick

Docket Nos. 50-324 and 50-325

License Nos. DPR-62 and DPR-71

Inspection at Brunswick site near Wilmington, North Carolina

Inspectors: Ross Butcher for D. F. Johnson, Senior Resident Inspector — 7-28-81 Date Signed

Ross Butcher for L. W. Garner, Resident Inspector — 7-28-81 Date Signed

Approved by: C. Julian, Acting Section Chief, Division of Resident and Reactor Project Inspection — 7/28/81 Date Signed

SUMMARY

Inspection on June 15 - July 15, 1981

Areas Inspected

This inspection involved 182 resident inspector hours on site in the areas of operational safety verification; review of operational events; review of periodic reports; follow-up on licensee event reports; plant tours; surveillance activities; follow-up on implementation of TMI task actions; observation of emergency drill table top sessions; follow-up on IEB's; follow-up on previous inspection findings; review and audit of on-site safety committee meetings; and independent inspection efforts.

Results

Of the 12 areas inspected, two violations were identified (failure to follow procedures, paragraph 8 and personnel error resulting in exceeding a limiting condition for operation, paragraph 9.)

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DETAILS

1. Persons Contacted

Licensee Employees

*C. Dietz, General Manager, Brunskwick
A. Bishop, Engineering Supervisor
G. Bishop, Project Engineer
*S. Bohanan, Principal Specialist Regulatory Compliance
J. Boone, Project Engineer
*J. Brown, Manager, Operations
J. Cook, E & RL Foreman
*C. Dietz, General Manager, Brunswick
J. Dimmette, Mechanical Maintenance Supervisor
M. Hill, Maintenance Manager
*R. Morgan, Plant Operations Manager
G. Oliver, E & RC Manager
A. Padgett, Assistant to General Manager
R. Poulk, Regulatory Specialist
W. Triplett, Administrative Manager
L. Tripp, RC Supervisor
W. Tucker, Technical and Administrative Manager

Other licensee employees contacted included technicians, operators and engineering staff personnel.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on July 14, 1981, with those persons indicated in Paragraph 1 above. Meetings were also held with senior facility management periodically during the course of this inspection to discuss the inspection scope and findings.

3. Licensee Action on Previous Inspection Findings

(Closed) Violation (325/81-06-01), removal of a component from 1-CAC-AQH-1262 unit that resulted in a violation of containment integrity due to inadequate procedural control. Appropriate review and training of this event was conducted for all cognizant personnel to prevent recurrence.

4. Reportable Occurrences

The below listed Licensee Event Reports (LER's) were reviewed to determine if the information provided met NRC reporting requirements. The determination included adequacy of event description and corrective action taken or planned, existence of potential generic problems and the relative safety significance of each event. Additional in-plant reviews and discussions with plant personnel, as appropriate, were conducted for those reports

indicated by an asterisk. LER's indicated with a double asterisk, remain open for one or more of the following reasons.

a. A supplemental report is required.
b. Items of work committed to for preventing recurrence, have not been completed.
c. A Plant Modification is required.
d. Equipment requires qualification per IEB 79-01B.
e. A Technical Specification Change is required.

Unit 1

**1-79-74 (3L)	Resin Injection into Reactor Vessel.
**1-79-97 (3L)	Loss of Emergency DG No. 1.
*1-80-57 (3L)	Standby Liquid Control Relief Valve 1-C41-F029B, did not lift.
*1-80-70 (3L)	No. 1 Diesel Generator declared inoperable.
*1-80-73 (3L)	RHR Flow Indicator Signal Converter 1-E11-FY-3338, Model No. SRT/4-20/1-5/45 STD, no response to input signal.
*1-80-74 (3L)	Inner Airlock Door Seals would not hold required pressure for specified length of time.
*1-80-75 (3L)	RPS Logic Relay, C71-K3E, did not de-energize, as required, when MSIV-F028B closed.
*1-80-80 (3L)	Failure of Reactor Low Water Level Switch 1-B21-LIS-N031A-2.
*1-80-83 (3L)	Failure of Containment Atmospheric Monitor Oxygen Analyzer 1-CAC-ATH-1259-2.
*1-80-85 (3L)	Failure of Reactor Low Water Level Switch 3 of 1-B21-LIS-N013B and D.
*1-80-87 (3L)	Rod Position Indication Failure.
**1-80-90 (3L)	Failure of Containment Atmospheric Oxygen Monitor 1-CAC-AT-1259
*1-80-92 (3L)	Reactor Vessel Conductivity over specifications.
*1-80-93 (3L)	Procedure deficiency relative to Rod Worth Minimizer.
*1-81-01 (1T)	Safety Relief Valve B21-F013G, Model 67F, malfunctioned.
**1-81-07 (3L)	Suppression Pool Level Transmitters 1-CAC-LT-2601 and 2602, improperly vented.
*1-81-09 (3L)	RCIC Oil Cooler Inlet Isolation Valve, 1-E51-F046, failed to open.
*1-81-10 (3L)	RCIC failed to accelerate above idle speed.
*1-81-11 (3L)	Recirculation Sample Inboard Isolation Valve, 1-B32-F019, Model T36D-1B, leakage.
*1-81-12 (3L)	Rod position indication malfunction.
**1-81-14 (3L)	Failure of HPCI Minimum Flow Valve F012 to close.
*1-81-17 (3L)	RCIC tripped off due to high steam exhaust pressure.
*1-81-02 (3L)	MSIV 1-B21-F028B Limit Switch failure.

*1-81-27 (3L)	Lockout on No. 1 Diesel Generator due to jarring of Potential Transformer.
**1-81-28 (3L)	Reactor Instrument Penetration Valve X-68-C, would not open.
**1-81-34 (1T)	Damaged supports on HPCI shaft broken.
1-81-40 (3L)	Required surveillance for ECCS Reactor Low Pressure Channel Calibration and Functional Test, PT 3.1.14 PC, not performed for ECCS Actuation Instruments, 1-B21-PS-N021B and D, by 5/24/81.
*1-81-47 (3L)	Primary Containment Isolation Valves, 1-CAC-V7, V9, V15, V16 and V17, failed local leak rate.
1-81-50 (3L)	3K snubber installed where design required 10K snubber on HPCI system 14 inch injection line piping.
*1-81-51 (3L)	Equipment hatches not verified to be closed.
*1-81-52 (3L)	RHR 1B Pump tripped.

Unit 2

2-80-19 (3L)	2B and 2D RHR SW Pumps could not meet required 4000 GPM at 300 PSIG discharge pressure.
2-80-26 (3L)	Main Steamline High Flow Instrument 2E41-DPIS-N004, found out of tolerance.
2-80-27 (3L)	Leak Detection Level Switch for HCU 26-47, did not actuate.
2-80-29 (3L)	E11-F049 Valve, RHR to Radwaste, could not be opened from the RTGB.
2-80-30 (3L)	2B RHR Heat Exchanger Rib Plate partially buckled.
2-80-31 (1T)	A ¼" throughwall horizontal crack located in heat-affected zone ¼" below weld on RWCU Isolation Valves F001 and F004.
2-80-34 (3L)	Snubbers 2G41-1SS22 on RHR suction from fuel pool cooling and 2SW-175SS166 on 2A RHR Heat Exchanger Discharge and 2E41-2SS105 on Discharge of HPCI, inoperable.
2-80-85 (3L)	CAD System Valves 2-CAC-V55 and V56 would not fully open, making CAD System inoperable.
2-80-100 (3L)	Rod Block Monitor RBM "A" Inoperable Alarm annunciated when-ever Rod associated with LPRM string 36-21C selected, caused by failure of RBM No. 5 first level Multiplexer card IC Chip, U7.
2-81-12 (3L)	Wiring deficiency associated with Actuation Switch 2-CAC-PDS-4223, prevented switch from operating as designed.
2-81-24 (3L)	B and D Main Steam Lines Leak Detection Actuation Relay Circuit Breaker mistakenly opened.
2-81-28 (1T)	Unit shutdown not completed within required 12 hour period.
2-81-44 (3L)	No. 4 Diesel Generator lockout received due to jacket water high temperature and low pressure trip signals.

2-81-48 (1T)	RIP Isolation Valve, X-206D, Position D, failed to open.
2-81-50 (3L)	Primary Containment Atmosphere Monitor Oxygen Analyzer 2-CAC-ATH-1259-2, Model No. F3, showed erratic indications of drywell and Suppression Pool Oxygen concentrations.
2-81-51 (3L)	No. 4 Diesel Generator declared inoperable.
*2-81-52 (3L)	2A Reactor Recirculation Pump declared inoperable.
2-81-53 (3L)	1" Drywell Nitrogen Inlet Isolation Valve, 2-CAC-V48, would not close within specified time limit, and deactivated.
2-81-54 (3L)	Drywell Floor Drain (DWFD) Sump Flow Integrator declared inoperable.
2-81-58 (3L)	Chart recorder of Primary Containment Atmosphere Monitor, 2-CAC-AR-1259, inoperable.

5. Review of Plant Operations

a. The inspector reviewed plant operations through direct inspections and observations throughout the reporting period. The following areas were inspected.

(1) Control Room
(2) Service Building
(3) Reactor Buildings
(4) Diesel Generator Rooms
(5) Control Points
(6) Site Perimeter

b. The following determinations were made:

-- Monitoring instrumentation: The inspector verified that selected instruments were functional and demonstrated parameters within Technical Specification limits.

-- Valve positions. The inspector verified that selected valves were in the position or condition required by Technical Specifications for the applicable plant mode. This verification included control board indication and field observation of valve position (Safeguards Systems).

-- Radiation Controls. The inspector verified by observation that control point procedures and posting requirements were being followed. The inspector identified no failure to properly post radiation and high radiation areas.

-- Plant housekeeping conditions. Observations relative to plant housekeeping identified no unsatisfactory conditions.

-- Fluid leaks. No fluid leaks were observed which had not been identified by station personnel and for which corrective action had not been initiated, as necessary.

-- Piping vibration. No excessive piping vibrations were observed and no adverse conditions were noted.

-- Control room annunciators. Selected lit annunciators were discussed with control room operators to verify that the reasons for them were understood and corrective action, if required, was being taken. Refer to paragraph 8, below for more details.

-- By frequent observation through-out the inspection period, the inspector verified that control room manning requirements of 10 CFR 50.54(k) and the Technical Specifications were being met. In addition, the inspector observed shift turnovers to verify that continuity of system status was maintained. The inspector periodically questioned shift personnel relative to their awareness of plant conditions.

-- Technical Specifications. Through log review and direct observations during tours, the inspector verified compliance with selected Technical Specification Limiting Conditions for Operation.

-- Security. During the course of these inspections, observations relative to protected and vital area security were made, including access controls, boundary integrity, search, escort, and badging. No unsatisfactory conditions were identified.

One violation was identified in this area. See paragraph 8.

6. IE Bulletin Followup

The following actions taken by the licensee were verified:

-- Written response was within the time period stated in the bulletin;

-- written response included the information required to be reported;

-- written response included adequate corrective action commitments;

-- information in response was accurate;

-- corrective action was described in the response;

-- copies of the response were forwarded to the appropriate on-site management representatives.

(Closed) IFI 50-324/81-BU-02 and 50-325/81-BU-02, Failure of Gate Type Valves to Close against differential Pressure: The Brunswick Plant has no

gate valves of the described manufacture in use or in stores. The licensee responded to the bulletin within 33 days instead of the required 30 days.

(Closed) IFI 50-324/81-BU-01 and 50-325/81-BU-01, Surveillance of Mechanical Snubbers: The Brunswick Plant has no mechanical snubbers in use. The licensee has committed to develop a mechanical snubber inspection program schedule if at some future time, this type of snubber is installed at Brunswick.

(Closed) IFI 50-324/80-BU-23 and 50-325/80-BU-23, Failures of Solenoid Valves Manufactured by Valcor Engineering Corporation. The Brunswick Plant uses neither valves nor valve parts manufactured by Valcor Engineering Corporation.

(Closed) IFI 50-324/80-BU-16 and 50-325/80-BU-16, Rosemount Pressure Transmitters: The Brunswick Plant neither uses nor plans to install the described transmitters in any safety-related application.

(Closed) IFI 50-324/80-BU-12 and 50-325/80-BU-12, Decay Heat Removal System Operability: Subject Bulletin is applicable only to PWR's.

(Closed) IFI 50-324/80-BU-09 and 50-325/80-BU-09, Hydramotor Actuator Deficiencies: The Brunswick Plant has no ITT General Controls Models AH-90 and NH-90, hydramotor actuators in use.

7. Surveillance Activities

The inspector witnessed portions of the following periodic test: PT 12.1.1 and PT 12.1.2A, Diesel Generator Actual Loading Test. These PT's verify that Unit 1 emergency buses shed loads and the diesel generators start and assume ECCS loads under simulated loss of off site power concurrent with a test ECCS actuation signal.

The inspector verified the following:

--Procedures conformed to Technical Specification requirements;
--procedure changes received proper review and approval;
--LCO's were met;
--test data is accurate and complete;
--test documentation was reviewed;
--test discrepancies were rectified;
--test results meet Technical Specification requirements;
--testing was completed by qualified personnel;
--surveillance schedule was met;
--system was restored to service;

No violations were identified in this area.

8. Control Room Annunciators

On April 21, 1981, the Plant Nuclear Safety Committee approved a new procedure OI-5-A, Status of Annunciators in Alarm. On the subsequent day, the procedure was signed by the General Manager and released to operations for implementation. On July 6, 1981, implementation of the procedure was inspected by the resident inspector. The results are provided below:

-- Step II.G. states, "This log will be used as a turnover item between shifts." Unit 2 shift turnover did not include review of the log. The Unit 2 control operator indicated that he was unaware of the requirement.

-- Eleven annunciators on Unit 2, panels 2-A-1 through 2-A-7, were alarming but were not recorded in the log.

-- Unit 1 and 2 control operators indicated that they were not maintaining the log as required.

-- Step II.D, requires a trouble ticket be initiated. Annunciator window 3-8 on panel 2-UA-23, 250V Batt B Ground, was alarming. Neither the control operator nor shift foreman knew on what prior shift the annunciator had come in, why it was alarming or whether a trouble ticket had been issued. A trouble ticket was issued after an investigation revealed that none had been initiated. The annunciator was not listed in the log.

These descrepancies constitute a Failure to Follow procedure. This is a Violation (50-324/81-14-02 and 50-325/81-14-02).

-- Thirty nine annunciators on the above-mentioned panels were not alarming, but listed in the log. The procedure makes no provision for recording in the date of entry or removal of annunciators from the log.

-- A shift foreman indicated that he was unaware of OI-5-A and the associated log. He had returned to control room duty approximately one week ago after assignment at another duty station.

-- A control operator indicated that operators did not have time to complete the log, as required, and had informed his supervisor of such.

-- No action had been taken to modify or cancel the procedure.

The preceding four items are Inspector Followup Item (50-324/81-14-03 and 50-325/81-14-03).

9. Personnel Error

a. While performing a normal reactor startup on June 29, 1981, the control room operator attempted to open the A-loop RHR torus suction valve

(F020A) remotely from the control room. After several attempts the valve failed to open.

General Plant Operating Procedure GP-1, Step B.2.3.3.1, requires that F020A be open prior to approach to criticality. The control room operator directed the auxiliary operator to open F020A manually from the reactor building. At 0528 hours, the control room operator placed the mode switch in startup and commenced a control rod withdrawal. At 0550 hours, it was discovered that F020A was not open. The auxiliary operator had encountered minor difficulties in manually opening the valve and the control room operator did not properly verify that valve F020A was open prior to commencing rod withdrawal. This is a violation of Technical Specifications 3.0.4 and 3.5.3.2, in that Section 3.0.4 prohibits entry into an operational mode unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the Action Statements. Section 3.5.3.2, Limiting Condition for Operation, requires two independent Low Pressure Coolant Injection subsystems of the residual heat removal system shall be operable with each subsystem comprised of:

(1) Two pumps.
(2) An operable flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

b. This event had relatively small significance regarding the health and safety of the public for the following reasons:

(1) The other redundant LPCI subsystem was operable;
(2) The backup alternate core spray subsystems were operable;
(3) The FSAR accident analysis takes credit for only one LPCI subsystem available during an accident condition;
(4) With the plant operating and one LPCI subsystem becomes inoperable, Power Operation may continue in accordance with Technical Specification action statement requirements provided both CSS subsystems are operable and the inoperable LPCI subsystem is restored within 7 days;
(5) The event was expeditiously terminated upon discovery of the position of the F020A valve.

This operator error in failing to meet Technical Specification requirements is a Violation (324/81-14-01).

10. Plant Transients

During the period of this report a follow-up on plant transients was conducted to determine the cause; ensure that safety systems and components functioned as required; corrective actions were adequate; and the plant was maintained in a safe condition.

a. On June 22, 1981, while Unit 2 was at 89% power, the 2A recirculation pump controller was nulled and the scoop tube was locked with flow bumping in progress at 1% every half hour. At 21.51 hours, the scoop tube was unlocked and recirculation pump speed immediately went to maximum. The reactor tripped on high flux, followed by a high water level trip and subsequent turbine trip. Group I isolation was reset and Procedure E1-31 "Reactor Scram" was carried out. All safety systems functioned as designed.

   The licensee in continuing to investigate this matter. No definite cause for this event has been determined. This is an inspector followup item. 50-324/81-14-04

b. On July 2, 1981, with Unit 2 at 96% power, the reactor tripped on low reactor water level at 0153 hours, the turbine tripped on reverse power. Subsequent testing of MSIV's revealed that the "C" main steam line was isolated when both inboard and outboard valves were open, indicating that 22C MSIV appears to have a separated disk. The cause of the trip was attributed to the sudden closing of MSIV 22C. This is the fifth event of this nature. Previous events and causes are described in I&E Inspection Report 81-12, paragraph 9.b.

   The inspector had no further questions relative to this event.

c. On July 2, 1981, a Unit 2 startup was in progress with the reactor at approximately 24% power. At 2009 hours, the reactor tripped and a Group I isolation was initiated by a loss of feedwater signal. A technician was valving in an isolated DP instrument following a calibration check when a pressure surge caused the closure of the RIP valve to the feedwater control.

   The inspector had no further questions relative to this event.

11. Review of Periodic Reports

The inspector reviewed the following Licensee Report.

-- Brunswick Steam Electric Plant, Units Nos. 1 and 2, Monthly Operation Report for May, 1981.

The inspector verified that the information reported by the licensee is technically adequate and satisfies applicable reporting requirements established in 10 CFR 50, and Technical Specifications.

The inspector had no further questions in this area. No violations were identified.

12. Onsite Review Committees

The inspectors attended the regular monthly Plant Nuclear Safety Committee (PNSC) Meeting and several special PNSC meetings conducted during the period of June 15 through July 15, 1981.

The inspectors verified the following items:

-- Meetings were conducted in accordance with Technical Specification requirements regarding quorom membership, review process, frequency and personnel qualifications;

-- Meeting minutes were reviewed to confirm that decisions/recommendations were reflected and follow-up of corrective actions were completed.

No violations were identified.

13. Office of the Analysis and Evaluation of Operational Data (AEOD) Site Visit.

A meeting was held on June 17, 1981, to discuss the recent incident involving the growth of oysters in the service water system and the resulting loss of RHR heat exchangers.

Personnel in Attendance:

Licensee Personnel

C. Benedict, Biological Lab.
C. Bohanan, Regulatory Compliance
J. Higley, Engineering
W. Hogarth, Manager, Environmental Technology Section
M. Hogle, Engineering
R. Morgan, Manager, Operations
G. Oliver, Manager, E&RC
N. Stalnaker, Chemist
W. Schade, Engineering
J. Titrington, Engineering
W. Tucker, Manager, Technical Support

NRC Personnel

J. Giannelli, AEOD
G. Imbro, AEOD
D. Johnson, Senior Resident I&E
M. Masnik, NRR

Agenda of the Meeting

I. Introductions

II. Chlorination

A. Program Description
B. Summary of Difficulties

III. 1981 Outage

A. Initial Symptoms and Actions
B. Discussion of Findings
C. Discussion of Actions
D. Biological Input - Prognosis

IV. RHR Heat Exchanger Specifics

A. Description of Problem
B. Repairs
C. Monitoring Program

V. Unit 1 Tour