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

Report Nos. 50-324/82-01 and 50-325/82-01

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, NC

Inspector: C. Julian for — 1/26/82
L. W. Garner, Resident Inspector — Date Signed

Approved by: C. Julian for — 1/26/82
C. Burger, Section Chief, Division of Resident and Reactor Project Inspection — Date Signed

SUMMARY

Inspection on December 15, 1981 - January 15, 1982.

Areas Inspected

The inspection involved 108 resident inspector hours on site in the areas of review of Licensee Event Reports, followup of plant trips, independent inspection, and operational safety verification.

Results

Of the 4 areas inspected, two violations were identified. (Failure to take reactor coolant samples per Technical Specifications 3.4.5.b.1, see paragraph 4.B, and failure to follow procedure per Technical Specification 6.8.1.a, see paragraph 4.c.).

DETAILS

1. Persons Contacted

Licensee Employees

A. Bishop, Engineering Supervisor
G. Bishop, Project Engineer
J. Boone, Project Engineer
*F. Coburn, Director QA/QC
*C. Dietz, General Manager, Brunswick
J. Dimmette, Mechanical Maintenance Supervisor
E. Enzor, I & C/Electrical Maintenance Supervisor
M. Hill, Maintenance Manager
*R. Knobel, Assistant Manager of Operations
*R. Morgan, Plant Operations Manager
*D. Novotny, Regulatory Specialist
*G. Oliver, E & RC Manager
R. Poulk, Regulatory Specialist
*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 January 11, 1982 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. Review of Licensee Event Reports

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.

Unit 1

1-81-64 (3L) Primary Containment Atmospheric Monitor Oxygen Analyzer, 2-CAC-AT-1263-2, tripped due to defective isolation valve.

*1-81-74 (3L) "B" RHRSW Subsystem declared inoperable.

1-81-75 (3L) Primary Containment Atmospheric Oxygen Analyzer, 1-CAC-AT-1263-2, out of calibration.

1-81-79 (3L) Input test signal to RHR system Flow Indicators, 1-E11-FI-3338, instrument exhibited full scale indication output.

1-81-82 (3L) Erroneous indication received for fully withdraw Control Rod 34-39.

1-81-86 (3L) Primary Containment Atmospheric Monitor Oxygen Analyzer, 1-CAC-AT-1259-2, tripped and would not remain running when restarted.

1-81-87 (3L) Primary Containment Atmospheric Monitor Oxygen Analyzer, 1-CAC-AT-1263-2, downscale indications of drywell oxygen concentration.

*1-81-88 (3L) Automatic isolating function of "C" Tip Machine Primary Containment Isolation Ball Valve, inoperable.

*1-81-94 (3L) Suppression Chamber Water Level Recorder 2-CAC-LR-2602, erroneous indications.

Unit 2

2-81-117 (3L) Primary Containment Atmospheric Monitor Oxygen Analyzer, 2-CAC-AT-1259-2, erroneous indication due to moisture build up in torus sample line.

*2-81-120 (3L) Actuation of "D" RHR Pump Discharge ADS Initiation Logic "A" Pressure Switch, 2-E11-PS-N016D, erroneous due to moisture accumulation.

2-81-121 (3L) Primary Containment Atmospheric Oxygen Analyzer, 2-CAC-AT-1259-2, high Drywell Oxygen concentration indicated due to moisture build up.

*2-81-123 (3L) "B" RPS Logic Relay, A71-K10B, Actuated due to an electrically grounded relay coil.

2-81-127 (3L) Primary Containment Atmospheric Monitor Recorder, 2-CAC--AR-1259, erroneous recorder input signal.

2-81-128 (3L) Primary Containment Atmospheric Monitor Oxygen Analyzer, 2-CAC-AT-1259-2, upscale indications of Drywell Oxygen concentration.

2-81-131 (3L) Primary Containment Atmospheric Monitor Hydrogen Analyzer, 2-CAC-AT-1263-1, slight Hydrogen concentration in Drywell.

*2-81-133 (3L) RTGB Instrument, 2-CAC-LR-2602, erroneous signal received for suppression chamber water level.

*2-81-134 (3L) Suppression Chamber Water Temperature Recorder, 2-CAC-TR-778, recorder print mechanism not recording.

4. Operational Safety Verification

The inspector verified conformance with regulatory requirements throughout the reporting period by direct observations of activities, tours of facilities, discussions with personnel, reviewing of records and independent verification of safety system status. The following determinations were made:

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

a. On December 31, 1981, at approximately 0815 hours while discussing work on B21-LT-N017D-2, a non-technical specification related instrument, the Unit 1 Shift Foreman was told that B21-LT-N017D-1 had a trouble tag attached to the meter referring to a trouble ticket, TT, written on December 28, 1981. The TT, 1-E81-4177, indicated that D-1 had failed upscale, the same condition D-2 was in. The Shift Foreman realized that D-1 was required to be operable per Technical Specification 3.3.1; however, no limiting condition for operation had been issued. After consultation with the shift operating supervisor, an LCO was taken out and, in apparent compliance with action statement 3.3.1.a, operators manually inserted a trip in channel B of the reactor protective system.

Trouble shooting by Instrument and Control, I&C, technicians revealed that the equalizing valve around D-1 was approximately 1/4 turn off its full close seat. It was postulated that the reference leg was partially empty. By use of a dead weight tester, about 1 and ½ pints of water was added to the reference leg. Both D-1 and D-2 returned to their normal indications. The LCO on D-1 was cancelled at 1330 hours on December 31.

Investigation into the event revealed the following:

1. Entries into the auxiliary operator's, AO, daily surveillance requirement, DSR, log revealed that D-1 had been reading

upscale, i.e., 210", since December 26, 1981. Readings for the previous two days had indicated an upward trend.

2. The AO DSR log provides no acceptance values for some data taken, e.g., readings on B21-N017D-1.

3. Shift turnover checklist OI-2, revision 6, requires the unit operator and the Shift Foreman to review the AO DSR log during each shift. These reviews from December 26 to December 31, 1981, failed to detect the abnormal reading.

4. A shift foreman, other than the one on December 31, had reviewed TT 1-E81-4177, checked that no LCO was required and initialed it. He told the inspector that it had been given to him about 5 minutes prior to the end of his shift and, because he had been informed during his shift turnover that D-2 was being worked, he either assumed or misread the TT to be on D-2.

5. In response to a question raised by the inspector on January 4, 1982, the licensee determined by January 6, 1982, that Technical Specification 3.3.2 had also been applicable and thus, because it had not been identified, the action statement 3.3.2.b requiring placement of the containment isolation actuation instrumentation channel in the tripped condition within 1 hour, had not been met. This Technical Specification refer to the instruments additional function of initiating a group 2, 6, 7 and 8 isolation on low water level.

6. Tripping of the reactor protection system, RPS, channel B, by pushing the manual scram button was not in literal compliance with Technical Specification 3.3.2.a, in that the effected instrument channel should have been tripped. If this had been done, both the RPS and the isolation actuation system would have been tripped.

7. After the plant modification which separated some instruments into "1" and "2", the operating staff had been taught that a "1" means RPS and "2" means isolation instrumentation. This general rule of thumb is incorrect for B21-LT-N017D-1 which serves both functions.

Inspection Report 81-24 contains a severity level IV violation in which an inoperable instrumentation channel, a HPCI temperature switch, was not placed in the tripped condition within 1 hour. In that instance, a shift foreman looked at the TT, assigned it a number and then left it to be completed later. The next shift processed the ticket, issued an LCO and completed the Technical Specification action statement. The similarity between that event on August 27, 1981 and the one on December 28, 1981, is that in both cases, TT's were handed the shift foreman close to shift change and both failed to closely scrutinize what had been handed then.

On January 6, 1982, B21-LT-N017D-1 and D-2 were noticed to be again trending upscale. The appropriate LCO's were written and Technical Specifications 3.3.1 and 3.3.2 were complied with by tripping the trip module associated with B21-LT-N017D-1. A small leak, approximately 11-12 drops per minute, was discovered at the packing gland nut on the flow check reset bypass valve at the penetration. The nut was tightened and the reference leg was refilled with water. The instrument appears to be working normally. The capacity of the condensing pot on the reference leg is estimated to be 4 gallons per hour. Very little water was evident around the leak. No puddles were seen by the inspector. Only some wetness on nearby items was observed. The apparent inability of the condensing pot to make up such a small leak is an Inspector Followup Item (325/82-01-01). A special inspection report number 82-02 is being issued which will include enforcement action on this event.

b. On December 18, 1981, in accordance with Technical Specification 3.4.5.b.2.a, a Unit 2 reactor coolant sample was taken at 2100 hours. The sample was counted at 2209 hours on December 18, and again at 1517 hours on December 19. In both cases, personnel failed to calculate the Dose Equivalent I-131. Calculation of the Dose Equivalent I-131 on December 20, from the data taken on December 18, revealed that the Dose Equivalent I-131 had been 0.266 uCi/gram. Technical Specification 3.4.5.b.1, requires sampling and analysis at least once per 4 hours until the specific activity is less than 0.2 uCi/gram Equivalent I-131. Failure to take the required samples is a violation (324/82-01-01).

Failure to complete the sample analysis was attributed to personnel setting the sample aside for later counting because I-132 was indicated as not meeting the summary criteria on the counting sheet.

Following the discovery of the failure to take the required samples a routine 3 hour sampling program was begun. At 0325 hours on December 21 dose equivalent iodine was found to be again within the Technical Specification limit.

On September 14, 1981, an event described in LER 2-81-99 occurred in which a technician failed to take a coolant sample within the prescribed Technical Specification time of 2 to 6 hours after a power increase. The sample was taken 1 hour and 43 minutes late. Corrective action included counseling all technicians on the importance of completing all applicable sampling and analysis requirements within a timely manner and changing the applicable RC&T procedures to better delineate Technical Specification requirements. These corrective actions were apparently not sufficient as demonstrated by the current event.

c. On December 30, 1981, review of the Unit 2 Shift Foreman's log indicated that the north RHR sump was being pumped to both the HPCI and south RHR sumps because the north sump pump was broken. Annunciator HPCI Room Door Open was lit. Discussion with the Unit 2 shift foreman revealed that both HPCI doors had been open since December 28, 1981. Annunciator procedure HPCI Room Door Open, revision 0, states: One door is allowed open if maintenance is in progress with shift foreman's permission. Failure to implement the procedure is a violation (324/82-01-02). One door was shut on December 30, 1981. Pumping of north RHR to south RHR sump was accomplished by routing the hose over the HPCI room instead of through it. The north RHR sump pump has since been repaired.

An Operating Experience Report Number 281-15, Unit 2 Service Water Spill to -17' Elevation, prepared by the BSEP engineering subunit, concluded that the October 7, 1981 spill would not have violated common mode flooding had spare pipe penetrations been sealed or the HPCI room doors been shut and watertight. See Inspection Report 81-27 for incident description. Futhermore, it recommended that the administrative controls be enforced as to keeping the HPCI doors closed. Implementation of this recommendation is an Inspector Followup Item (324/82-01-03).

Additional review of the log indicated that at 0300 hours on December 22, 1981, a piping connection at the -17' level between the north core spray (CS) room and north RHR room, which is normally blank flanged closed, was opened to allow pumping the north RHR sump to the north CS sump. The opening of this penetration is outside the scope of the October 6, 1972 Reactor Building Flooding Summary Report prepared for the licensee by United Engineers & Constructors, Inc. The potential loss of redundancy due to common mode flooding between north CS and north RHR systems is a matter addressed in the Brunswick Final Safety Analysis Report, Appendix M, Response M6.33-1. That report states that the pipe chases between the core spray and RHR rooms are sealed to prevent flooding. It also states that the RHR and HPCI pump rooms are separated by walls up to elevation +5 feet for flood protection. That is only true when the HPCI room doors are closed. Together these events are a deviation from commitments made to the NRC. (324/82-01-04). Operation with both HPCI room doors open has been a recurring problem and the result of previous NRC enforcement action at Brunswick. Additional information is being gathered by the licensee about the circumstances and remedial actions implemented, if any, during the period the line was open.

d. By observation during the inspection period, the inspector verified the 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.

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.

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

Safeguard system maintenance and surveillance: The inspector verified by direct observation and review of records that selected maintenance and surveillance activities on safeguard systems were conducted by qualified personnel with approved procedures, acceptance criteria were met and redundant components were available for service as required by Technical Specifications.

Major components: The inspector verified through visual inspection of selected major components that no general condition exists which might prevent fulfillment of their functional requirements.

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

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.

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

Radioactive releases: The inspector verified that selected liquid and gaseous releases were made in conformance with 10 CFR 20 Appendix B and Technical Specification requirements.

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.

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.

Two violations and one deviation were identified in this area.

5. Followup of Plant Transients and Safety System Challenges

During the period of this report, a followup on plant transients and safety system challenges 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.

On December 18, 1981 at 1704 Unit 2 reactor experienced a scram from approximately 7% of full power when the mode switch was changed from run to startup. Main steam isolation valves, MSIV's, closed and attempts to open them were not successful until about 1900. Until then reactor pressure and water level was controlled by manual operation of safety relief valves and operation of the reactor core isolation cooling, RCIC, system and the High Pressure Coolant Injection, HPCI system. After 1900, the main condenser was used as a heat sink and cooldown proceeded normally. During the event reactor pressure did not exceed 1040 psig and vessel level remained above 150 inches.

Investigation revealed that malfunctioning pressure switches on the main steam lines had indicated steam flow greater than 40% with the mode switch in startup. Thus closure of the MSIV's had occurred. The reactor then scrammed on high reactor pressure and flux greater than 15% when in startup. Difficulty in reopening the MSIV's was also attributed to these same switches. The switches were repaired and procedural changes are being implemented to verify these are not tripped prior to changing the mode switch in future power reductions.

During the event, RCIC, tripped on high vessel level and could not be restarted. The solenoid which allows the trip and throttle valve to be relatched had shorted to ground. Repairs have been completed.

The inspector has no further questions about this event at this time.