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

Report Nos. 50-324/82-08 and 50-325/82-08

Licensee: Carolina Power and Light Company
411 Fayetteville Street
Raleigh, N. C. 27602

Facility Name: Brunswick 1 and 2

Docket Nos. 50-324 and 50-325

License Nos. DPR-62 and DPR-71

Inspection at Brunswick site near Wilmington, N. C.

Inspectors: C. Julian for — 4/2/82
L. W. Garner, Resident Inspector — Date Signed

C. Julian for — 4/2/82
J. Chase, Resident Inspector, Browns Ferry — Date Signed

C. Julian — 4/2/82
C. Julian, Project Inspector — Date Signed

Approved by: [signature] — 4-9-82
C. W. Burger, Section Chief, Division of Project and Resident Programs — Date Signed

SUMMARY

Inspection on February 15 - March 15, 1982

Areas Inspected

This inspection involved 225 inspector hours on site in the areas of followup of violations, review of Licensee Event Reports, review and audit of onsite safety committee meetings, review and audit of maintenance activities, followup of plant transients and safety system challenges, operational safety verification, followup on TMI Task Action Plan items, review and audit of surveillance activities and independent inspection.

Results

Of the 9 areas inspected, one violation with two examples was identified, (Failure to take proper corrective action, see paragraph 3) and one deviation was identified (Failure to revise procedures in accordance with TMI Action Item requirements, see paragraph 10).

DETAILS

1. Persons Contacted

Licensee Employees

A. Bishop, Engineering Supervisor
J. Boone, Project Engineer
*C. Dietz, General Manager, Brunswick
J. Dimmette, Mechanical Maintenance Supervisor
*W. Dorman, QA Supervisor
E. Enzor, I&C/Electrical Maintenance Supervisor
M. Hill, Maintenance Manager
R. Knobel, Manager of Operations
D. Novotny, Regulatory Specialist
*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 March 12, 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. Licensee Action on Previous Inspection Findings

(Closed) Violation (324, 325/81-19-01) Failure to maintain control copies of procedures up to date. The inspector performed a spot check of operating procedures and emergency instructions in the control room and found no discrepancies. The inspector was informed that Quality Assurance is also performing a quarterly surveillance of these procedures. The inspector had no further questions.

(Closed) Violation (324, 325/81-19-02) Failure of personnel to follow posted requirements on Radiological Work Permit (RWP). The inspector verified that all personnel at the site were instructed at the November, 1981, monthly safety meeting to observe all requirements on RWP's. In addition, the inspector observed that instructions were properly posted on how to don anti-contamination clothing. The inspector had no further questions.

(Open) Violation (324/81-20-03, 325/81-20-02) Uncontrolled release of liquid waste. The licensee's response to this item, dated October 12, 1981, stated that "E&RC Procedure 2010 will be revised to require two independent line up verifications on the radwaste control panel prior to the release of the Floor Drain Sample Tanks (FDST's), Waste Sample Tanks (WST's) and Detergent Drain Tanks (DDT's)." On March 6, 1982, the inspector observed the discharge of the DDT and noticed that no second verification was performed. A review of E&RC Procedure 2010, showed that second verifications were only being performed on the FDST and WST to ensure that the correct pump had been started. No second verifications were being performed on the radwaste panel prior to a release on the FDST's, WST's or DDT's.

Failure to perform a second verification prior to the release of the DDT's, FDST's and WST's is an example of an apparent violation for failure to take adequate corrective action (324, 325/82-08-01).

(Open) Violation (324/81-20-04, 325/81-20-03, 324/81-28-02 and 325/81-28-02) PNSC review of temporary change greater than 14 days. The inspector reviewed the licensee's response to this item which stated that, "A person in the Operations Unit will be tasked with tracking of all temporary revisions and ensuring that they are properly reviewed in the required time frame." The inspector's review of this item showed the following problems:

a. The person tracking temporary revisions had no formal instructions on the method of tracking.

b. The person tracking temporary revisions was informed to track temporary revisions until they reached the PNSC and not until they are signed by the General Manager. The General Manager must sign all temporary revisions within 14 days of implementation as required by TS 6.8.3. Not tracking the temporary revision until it was signed by the General Manager resulted in not having a temporary revision to the Auxiliary Operator Daily Surveillance Report reviewed and approved within 14 days. The licensee identified to the resident inspector, the review process for this item took 22 days.

Failure to develop an adequate tracking system for temporary revisions to procedures is an example of an apparent violation for failure to take adequate corrective action (324, 325/82-08-01).

(Closed) Inspector Followup Item (324, 325/81-29-02) Shift turnover checklist to be revised. This item was opened by the resident inspector because the shift turnover checklists did not specifically designate those items that must be performed prior to assuming shift duties nor did it designate responsibilities. The inspector reviewed the revised shift turnover checklists and had no further questions.

(Closed) Unresolved (325/81-29-03) Apparent failure to protect safety-related equipment during maintenance. The inspector toured the Reactor Building and observed work in progress in regards to equipment protection. The inspector also held discussions with craftsmen at the job location in regards to protection of equipment. The inspector had no further questions in this area.

(Closed) Unresolved (324/81-11-01, 325/81-11-02) Failure to post radiation levels for drywell. The inspector reviewed Radiological Control Procedure 0250, "Posting of Areas/Materials", which showed the procedure had been revised to include requirements to periodically update the drywell area surveys at the drywell. The inspector did not actually observe the implementation of this item since both drywells were secured. The inspector had no further questions in this area.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. New unresolved items identified during this inspection are discussed in paragraphs 5 and 7.

5. 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

1-81-089-3L	Annunciator indicating high suppression pool level not recognized by on shift operator.
1-82-04-3L	Post-accident monitoring control room recorder/ indicator, 1-CAC-AR-1259, exhibited unvarying indication of drywell oxygen concentration.
*1-82-07-3L	RCIC turbine control valve, 1-E51-V9, would not fully open or close, preventing proper speed control of RCIC turbine.
*1-82-010-3L	Operator fails to request dose equivalent I-131 determination after power change. Corrective action to prevent recurrence will be reviewed during a future inspection.
*1-82-011-3L	Reactor recirculation pump 1B trips due to improper training of Instrument and Control technicians Corrective action to prevent recurrence will be reviewed during a future inspection.

*1-82-14-3L	Unit No. 1 Post-accident monitors, 1-CAC-AQH-1260-1 and 2, 1261-1, 2 and 3 and 1262-1 and 2 showed upscale indications.
1-82-15-3L	Accumulator low pressure/high level alarm annunciator received for hydraulic control unit (HCU) to control rod 38-47 and control rod declared inoperable.
1-82-16-3L	Containment atmosphere oxygen monitor, CAC-AT-1263-2, would not calibrate, due to burned out monitor exciter lamp.
*1-82-093-1T	Reactor protection system and primary containment isolation system channels not tripped when instrument fails non-conservatively. Corrective action to prevent recurrence will be reviewed as part of inspection followup of violation 325/82-02-01.

Unit 2

*2-80-85-3L	CAD system valves 2-CAC-V55 and V56 would not fully open making CAD system inoperable.
*2-81-118-3L	Secondary containment fire seals fail periodic visual and leak test.

The inspector reviewed LER 2-81-118, secondary containment fire seals failing Periodic Test (PT) 35.16.2. The licensee reported five fire seals failing the PT, however, the inspector determined, and it was confirmed by the licensee, that seven seals had actually failed to pass the PT. The licensee has agreed to resubmit the LER.

The inspector also questioned the method the licensee was using to count failed penetrations. The licensee was counting as one penetration segments in which up to 50 or more penetrations could penetrate the segment. This was discussed with the Region II fire protection inspector and the licensee. An agreement was reached in which the licensee would continue to count penetrations as they had previously been doing but, would also report how many individual penetrations in a segment failed.

The inspector also reviewed PT 35.16.2, Fire Barrier Penetration Seals Reactor Building (Unit 2). The review of the completed procedure showed that seven penetrations had not been performed and the PT was signed off as complete, with no rescheduling necessary. Therefore, the PT was not scheduled to be performed again until 1983. This PT is required to be performed every 18 months and these seven penetrations were last done on August 25, 1980, therefore, they have to be performed by June 1982. By not performing the PT again until 1983, the licensee would have violated Technical Specification 3.7.8 in June of 1982.

This PT was reviewed by three individuals, including a QA review, after completion of the PT and was signed off as not needing rescheduling. Failure to reschedule the PT is an example of an apparent weakness in the review process to be reviewed further. This item is unresolved (324, 325/82-08-11)

*2-81-119-3L	RHR relay logic "A" power failure occurred making LPCI function of RHR inoperable
*2-81-132-3L	Reactor water cleanup isolation valve motor power fuses blew.
*2-81-143-3L	Nordberg diesel generator, FS-1316-HS6, experiences erratic load control.
*2-82-2-3L	Suppression chamber water level indicator, 2-CAC-LI-3342, located on remote shutdown panel, out of calibration.
*2-82-004-3L	Frozen condensate in containment atmospheric control system makes vaporizer inop.

During the inspector's review of LER 2-82-4, the inspector determined that there was no formal method for inspecting lines requiring freeze protection prior to the onset of winter. The licensee has committted to developing a checklist to identify those lines needing to be inspected for proper freeze protection by winter of 1982 - 1983. (IFI 324, 325/81-08-02)

*2-82-007-3L	Reactor coolant activity exceeded tech spec limits.
*2-82-009-IT	Fire watch not established in service water building when sprinkler system inopt.
	Corrective action to prevent reoccurrence will be reviewed.
*2-82-13-3L	Reactor low level switch 2-B21-LIS-N07D-1, inaccurate instrument response to level changes.
*2-82-15-3L	IRM "E" indicated downscale when on range 9 and IRM "C" previously declared inoperable due erratic indication.
*2-82-009-1T	Fire watch not established in service water building when sprinkler system inoperable.
*2-82-13-3L	Reactor low level swtich 2-B21-LIS-N07D-1, inaccurate instrument response to level changes.

*2-82-15-3L	IRM "E" indicated downscale when on range 9 and IRM "C" previously declared inoperable due erratic indication
*2-82-21-3L	Discrepancy between narrow and wide range instruments of suppression chamber water level indications on RTGB.
*2-82-026-3L	Automatic depressurization system actuation switch B21-LIS-N031B-2 fails to actuate.
*2-82-003-3L *2-82-025-3L *2-82-027-3L	Generic failure of GE type HFA relays at Brunswick. Melted insulation from relay coils caused relays to fail to energize or deenergize. This was identified in General Electric Service Information Letter No. 44 and IE Notice 81-01. The licensee has committed to monthly inspect the continuously AC energized HFA relays until they can be replaced. Replacement of the affected relays by the first outage of sufficient length after receipt of an improved type is an inspector followup item (324/82-08-03 and 325/82-08-03).

6. Onsite Review Committee

The inspectors attended several special Plant Nuclear Safety Committee (PNSC) meetings conducted during the period of February 15 through March 15, 1982.

The inspectors verified the following items:

- 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 and follow-up of corrective actions were completed.

No violations were identified.

7. Review and Audit of Maintenance Activities

Certain Q List items were found to be incorrectly specified in Maintenance Instructions, MI, as Non-Q. On January 20, 1982 review of MI 3-3A34, SW-PS 1175 and 1176 service water pressure switch D2T-M80SS-L6, revisison 0 dated December 26, 1979, disclosed that the MI specified that these switches are not Q-list items. On February 18, 1982 review of MI 3-A29, C71/C72-PS-N003A-0 Barksdale Pressure Switch Model P1H-M340SS, revision 1 dated October 29, 1979, disclosed that the MI specified that these switches are not a Q list item.

In both cases Operating Manual Vol. XI Book 2 revision 16 dated December 4, 1981, list these as Q in Table I but does not list them as Q in Table IA. Table IA is a computer generated list used by maintenance personnel as a quick reference. On March 3, 1982, the licensee issued a revision to Table IA so that this and other similar discrepancies between the Tables are resolved. Whether or not these switches were procured and installed as Q items is an unresolved item pending additional review by the inspector. (324/82-08-04 and 325/82-08-04)

Furthermore, it was noted that MI 3-A29 specifies that the N003 switches be calibrated 'as required'. Review of Unit 2 calibration records indicates that they were last calibrated on March 27, 1978. The N003 switches are used to allow bypasses of the turbine control valve and stop valve trips in the RPS circuitry when reactor power is less than 30% as measured by these switches. The licensee has committed to revise MI 3-A29 to require an annual calibration of these switches. This is an Inspector Followup Item (324/82-08-05 and 325/82-08-05).

8. 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.

A. Unplanned Engineered Safety Feature (ESF) Initiations

1. On February 11 and 12, 1982, Unit 1 reactor had a series of inadvertent ESF actuations while operating at 8% of full power. These occurred at 1951 hours on February 11 and at 0053, 0102 and 0113 hours on February 12. Each of these events caused both core spray pumps A and B and all four diesel generators to auto start. In addition recirculation pump 1B tripped off and a half reactor scram and half primary containment isolation system, PCIS, group 1 (Main steam line isolation valves, MSIV, closure) were received on channel B. No injection of water into the vessel occurred. The first event also caused a partial High Pressure Coolant Injection, HPCI, initiation. The HPCI auxiliary oil pump started but the steam valve to the turbine did not open. After each event the associated ESF equipment was placed in standby and the recirculation pump 1B restarted.

   Investigation after the first three events revealed that the fuses in the circuit supplying power to the reactor protection system, RPS, and emergency core coolant system, ECCS, channel B cabinets had blown. Loss of power to these cabinets have caused similar events in the past. See Inspection Reports 325/81-20, 81-24 and 81-31. The initiating cause for the fourth event was not momentary in nature. The battery charger for battery B2 was found to have failed so that the output voltage indication was full scale, greater than 150 volts, and some cells were boiling over.

The battery charger was removed from service. The probable cause for the high voltage spikes and ultimate failure of the battery charger voltage regulator circuit was determined to be a non-soldered connection.

In response to previous events, the licensee had begun development in November 1981 of a modification which would sense a high battery charger output voltage and trip the battery charger input breaker. During February 5 and 8, 1982, a special test procedure, SP-81-47, 125 VDC Battery Charger Surge Test was successfully performed to demonstrate the ability of the modification to terminate a high voltage spike prior to the RPS and ECCS cabinet circuit protection actuating. Thus after plant modifications 82-030 and 82-031 are installed no future similar occurrences are anticipated. Installation of these modifications by the next refueling outages is an Inspector Followup Item (324/82-08-06 and 325/82-08-06). The Licensee has also examined all the battery chargers for this and other similar manufacturing defects.

The inspector reviewed the logic prints for HPCI, Core Spray and Low Pressure Coolant Injection, LPCI, mode of the residual heat removal system to determine why the ESF systems responded partially as they did to the events. The following conclusions were reached.

a. The transient trip signal for low low low water level was of sufficient duration that the seal-in relays in the core spray pump start circuits were energized. These seal-in relays are in series with another set of relays which must actuate first.

b. The transient trip signal for low low low water level was not of sufficient duration for the seal-in relays of the LPCI pump start circuits to actuate. These seal-in relays require two sets of other relays to energize prior to them actuating.

c. The starting circuit for the diesel generators is similar to that of the core spray pumps.

d. The core spray injection valves and other similar large valves did not take their accident positions. The power transient most likely caused all permissive conditions to be momentarily satisfied. However no valve motor energization occurred because the seal-in function requires the motor contactor to energize. The time required for this to happen is several times that of the response time of the smaller relay coils associated with the pump start circuits.

e. The steam supply valve to the HPCI turbine did not open for the same reason as item d above.

The inspector has no further questions at this time.

2. On February 19, 1982 at 2207 hours while shutdown Unit 1 reactor experienced injection of water into the vessel by both core spray pumps and B loop of the Residual Heat Removal (RHR) system. Vessel level increased approximately 40" before the pumps were shutdown. The diesel generators also auto started as expected.

   Prior to the event, the B21-LTM-N031A, B, C and D instruments were observed to be reading full scale, greater than 210". The acceptance criteria in the daily surveillance log was for power operation. The criteria had recently been added in response to a violation 325/82-02-01. However, because operating personnel were unsure as to whether this was a normal reading during shutdown or an indication of a non-full reference leg, an Instrument and Control (I&C) technician was sent to fill the reference leg. Overpressurization of the reference leg forced N031 B and D downscale below the vessel low low low level trip setpoint. All ESF system components except RHR pump A responsed as expected.

   Loop A of RHR was in shutdown cooling mode with C RHR pump operating. RHR pump A started and operated approximately 90 seconds with its suction valves closed. The pump interlock to prevent operation with the suction valves closed malfunctioned. The licensee has issued a trouble ticket to correct the malfuction. The licensee has committed to develop shutdown acceptance criteria for the daily surveillance log without field testing unless absolutely necessary.

   The inspector had no further questions at this time.

B. Turbine Control Valve Closure Trips Unit 1

On February 18, 1982 at 1154 hours, Unit 1 reactor experienced a scram from 80% of full power as a result of turbine control valve (TCV), No. 2 going full closed. Prior to the scram, TCV No. 2 has stuck at the 95% open position. During the attempt to free it, the valve went shut. All four bypass valves opened, however their response was not rapid enough to prevent a high reactor pressure trip on reactor protection system, RPS, channel B. This combined with the TCV fast closure trip of RPS channel A due to the TCV No. 2 closure, initiated the reactor scram

A main steamline isolation, Group 1, occurred and both High Pressure Coolant Injection, HPCI, and Reactor Core Isolation Cooling, RCIC, systems auto started. RCIC tripped on overspeed but was manually restarted to control vessel level. The Group 1 was reset at 1209 hours and normal cooldown commenced. HPCI did not inject because the E41-F006 valve did not open. Investigation by the licensee concluded that the low level initiation signal had not been present long enough for the HPCI turbine stop valve full open permissives to the E41-F006 valve to occur simultaneously. An injection test was run at 1610 hours to verify the operability of HPCI.

Subsequent testing of RCIC resulted in several high turbine exhaust pressure trips. On February 23 the probable cause of these trips as well as the overspeed trip of RCIC, was determined to be the ramp time for the ramp generator was too fast. The setting was adjusted from 9 seconds to 20 seconds. RCIC was subsequently tested and performed normally.

The narrow range reactor pressure strip chart recorder indicated that reactor pressure did not exceed 1050 psig during the transient.

The inspector has no further questions at this time.

C. Feedwater Transient Causes Low Level Scram

On March 13, 1982 at 1901 hours, Unit 2 reactor experienced a low water level scram from 64% of full power. Prior to the event, reactor feedwater pump, RFP 2A was being placed in service and RFP 2B was to be removed from service. The recirculation valve on RFP 2A went from shut to full open. RFP 2B was unable to respond rapidly enough to prevent a flow decrease to the vessel. Reactor vessel level decreased to approximately 170 inches and a reactor low water level scram occurred. The High Pressure Coolant Injection system was started but not required for vessel level control. The RFP supplied makeup to the vessel. Calibration of the B21-LTM-N017 switches revealed that one in each low water level trip channel had its setpoint drift up above 170 inches.

The inspector has no further questions at this time.

9. 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:

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

b. Operator performance: 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.

c. 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.

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

e. 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.

f. 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.

g. 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).

h. 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.

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

j. 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.

k. 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.

   1. The inspector followed up on two spills of condensate water from the temporary demineralizer used to reduce the levels of organics in the Condensate Storage Tank. The spills, which occurred on March 2 and 3, 1981, resulted in approximately six gallons of potentially contaminated water being released to the ground on each spill.

      The plant Radiological Control Unit took soil samples on each spill to determine if any activity was released. The results of these samples are as follows:

March 2, 1981	One sample	7.7 E-6 uci/cc
March 3, 1981	First sample	7.7 E-6 uci/cc

Second sample	1.87 E-5 uci/cc
Third sample	1.4 E-4 uci/cc

The inspector's review of this incident showed that on March 3, 1981, only portions of the spill area were posted as a potentially contaminated area. The entire area was not posted until after the third sample and the inadequate posting was brought to the Radiological Control Unit Supervisor's attention by the inspector.

As a result of this incident, CP&L has committed to:

a. Not using the temporary demineralizer until the high level alarm is repaired and a temporary dam is erected to contain any spillage.

b. A watch will be stationed at the temporary demineralizer at all times, while it is in use, to secure the system if spills occur.

c. Health Physics personnel will be instructed by March 15, 1982 to post any suspected areas as potentially contaminated until survey results prove otherwise (324, 325/82-08-07).

2. On March 10, 1982, the inspector observed an individual removing Anti-C clothing during General Employee Training, G.E.T. One step-off pad was being used. Because it is not abnormal for two step-off pads to be at the exit of contaminated areas, the inspector requested the licensee to consider using two pads during the Anti-C dress out portion of G.E.T. This is an Inspector follow-up Item (324, 325/82-08-08).

10. Followup on TMI Action Items

Item II.B.4 - Training for Mitigating Core Damage.

The licensee has completed training required for Shift Technical Advisors and operating personnel. Although a required retraining frequency is yet to be established, the licensee is proceeding to factor mitigation of core damage material into the normal requalification program. This item is closed.

Item II.E.4.1 - Dedicated Hydrogen Penetrations

The licensee is currently committed to install the dedicated hydrogen penetrations for recombiner hookup on both units during the 1982 refueling outage. Licensee representatives stated that due to the uncertainty of the impact of a proposed change to 10 CFR 50 requiring all plants to have recombiner capability, the dedicated penetration installation has been deferred. The inspector stated that the licensee should promptly inform NRR in writing of this proposed change to their commitment. This item remains open.

II.E.4.2.6 - Containment Purge Valves

The licensee has performed plant modifications to limit the stroke of vent and purge valves and closed under clearance those that could not be modified. A series of modifications have been completed to modify the isolation override logic to obtain diversification. Modifications have been completed to the reset logic to prevent automatic reopening upon resetting isolation signals. The qualification study for Brunswick purge and vent valves has been submitted to NRR by letter of November 17, 1981. NRR issued a SER on December 15, 1981 stating that routine purging should be accomplished through one half inch lines to standby gas treatment. CP&L responded by letter of 2/12/82 stating that purging through such a small line is not practical and stated that a further response will be submitted within six weeks. As appropriate modifications have been completed and qualification data submitted, this item is closed.

II.E.4.2.7 - Radiation Signal Closing of Purge Valves.

Following some confusion over the exact meaning of this requirement, CP&L has come to understand that the requirement is to install instrumentation to isolate purge and vent valves upon a high radiation signal within the drywell. By letter of June 30, 1981, CP&L stated to NRR that such an isolation is unnecessary and they do not intend to install one. CP&L representatives state that their position is a common one with the BWR Owners Group. CP&L further states that additional justification will be submitted to support their position. As no modifications are planned, this item is closed.

II.K.3.13 - Separation of HPCI and RCIC Initiation Levels.

CP&L has adopted the common response of the BWR Owners Group which says there is no benefit to initiating HPCI and RCIC at different reactor vessel levels. A letter from General Electric to NRR dated October 1, 1980 presents analysis supporting this conclusion.

CP&L has completed modifications to both units to allow RCIC to auto-reset following isolation on high vessel level. This is accomplished by closing the RCIC steam supply valve on high level rather than the trip and throttle valve. This item is closed.

II.K.3.15 - Modification to HPCI and RCIC Steam Line Break Isolation Logic.

By letter of December 31, 1980 CP&L presented the common BWR Owners Group response to this question. The recommended fix of installing a 3 second time delay in the isolation logic such that a high steam flow signal must be present for 3 seconds before isolation will occur has been adopted. Modifications were completed for HPCI and RCIC on both units as of June 2, 1981. This item is closed.

II.K.3.18 - Modification of ADS Logic.

By letter of April 22, 1981 CP&L submitted to NRR the BWR Owners Group analysis of the feasability of modifying ADS logic to allow automatic ADS initiation without high drywell pressure as a prerequisite. CP&L stated that no significant benefits would result from the modification and states that no modifications will be made. This item is closed.

II.K.3.19 - Recirculation Pump Interlock.

Brunswick is a jet pump design and this item is not applicable. This item is closed.

II.K.3.14 - Isolation Condenser High Radiation Isolation.

Brunswick has no isolation condenser and this item is not applicable. This item is closed.

II.K.3.27 - Common Reference Level for Vessel Level Instrumentation.

Modifications have been physically implimented on both units to adopt a common reference level. As of 2/26/82, all documentation has not been completed on Modification Package 80-180 which installed this system on Unit 1. Review of the completion of this Modification Package documentation will be reviewed during a future inspection and is an inspector followup item (50-325/82-08-09). This item is closed.

II.B.3 - Post Accident Sampling.

The licensee has committed to begin installation of the improved sampling station outside the reactor buildings during the 1982 refueling outages. This item remains open.

I.A.1.3 - Shift Manning

In response to this item CP&L stated in letters of November 5, 1980 and December 15, 1980 to the NRC that their procedures are in conformance with Darrell G. Eisenhut's letter July 31, 1980, which requires at least one licensed senior reactor operator (SRO) in the control room at all times, other than cold shutdown. Inspectors reviewed the administrative procedures that specify the control room manning requirements and found that they do not specifically require a SRO to be in the control room at all times other than cold shutdown. This is a deviation from commitments made to the Commission (50-324, 325/82-08-10). The licensee promptly modified their procedures to meet the requirements. Licensee representatives stated that it has been the practice to have on SRO in the control room, even though procedures do not specifically require it.