UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report No.: 50-400/93-12

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket No.: 50-400 Licensee No.: NPF-63

Facility Name: Harris 1

Inspection Conducted: May 15 - June 18, 1993

Inspectors: R E Carroll Jr for 7/1/93
J. Tedrow, Senior Resident Inspector Date Signed

R E Carroll Jr for 7/1/93
D. Roberts, Resident Inspector Date Signed

Approved by: H. O Christensen 7/2/93
H. Christensen, Chief Date Signed
Reactor Projects Section 1A
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted by two resident inspectors in the areas of plant operations, radiological controls, security, fire protection, review of nonconformance reports, surveillance observation, maintenance observation, safety system walkdown, followup of onsite events, outage activities, adverse weather operations, licensee event reports, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed. Some of these tours and observations were conducted on backshifts.

Results:

Three violations were identified: Failure to maintain an electrical containment penetration conductor overcurrent protection device operable, paragraph 2.c.(1); Failure to properly implement plant procedures for the control of locked valves and for the operation of electrical switchgear, paragraphs 2.c.(3) and 6; and Failure to verify design accuracy of auxiliary feedwater recirculation piping, paragraph 10.

Weaknesses were noted in the areas of scaffolding control, paragraph 2.b.(7) and in the documentation of a boric acid leakage inspection, paragraph 7.

More problems were experienced with 480 volt Asea Brown Boveri LK Circuit breakers, paragraph 2.c.(2).

Good pre-planning and use of equipment mockups were evident for containment work, paragraph 7. In addition, the simulator was utilized to provide plant startup/shutdown training and potential transients. Also, simulator training was provided for the conduct of an infrequent turbine test, paragraph 3.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

D. Batton, Manager, Work Control
J. Collins, Manager, Training
*C. Gibson, Manager, Programs and Procedures
M. Hamby, Manager, Regulatory Compliance
D. McCarthy, Manager, Regulatory Affairs
T. Morton, Manager, Maintenance
*J. Moyer, Manager, Site Assessment
J. Nevill, Manager, Projects
*W. Robinson, General Manager, Harris Plant
*W. Seyler, Manager, Outages and Modifications
H. Smith, Manager, Radwaste Operation
*D. Tibbitts, Manager, Operations
G. Vaughn, Vice President, Harris Nuclear Project
*B. White, Manager, Environmental and Radiation Control
W. Wilson, Manager, Spent Nuclear Fuel
*L. Woods, Manager, Technical Support
*A. Worth, Principal Engineer, Nuclear Engineering

NRC Representatives

*J. Johnson, Deputy Director, Division of Reactor Projects

Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation and corporate personnel.

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Review of Plant Operations (71707)

The plant began this inspection period in power operation (Mode 1). At 12:07 a.m., on May 22, 1993, a plant shutdown was performed to repair steam generator blowdown valve 1BD-8 located inside containment. The plant was placed in the hot standby condition (Mode 3) at 12:44 a.m. to effect repairs. Following completion of these repairs, a reactor startup was commenced and criticality achieved at 3:19 a.m., on May 22. Power operation was resumed at 1:30 p.m., on May 23. The plant continued in power operation for the remainder of this inspection period.

a. Shift Logs and Facility Records

The inspector reviewed records and discussed various entries with operations personnel to verify compliance with the Technical Specifications (TS) and the licensee's administrative procedures.

The following records were reviewed: shift supervisor's log; control operator's log; night order book; equipment inoperable record; active clearance log; grounding device log; temporary modification log; chemistry daily reports; shift turnover checklist; and selected radwaste logs. In addition, the inspector independently verified clearance order tagouts.

The inspectors found the logs to be readable, well organized, and provided sufficient information on plant status and events. Clearance tagouts were found to be properly implemented. During a review of the shift supervisor's log, the inspector noted an entry for June 7, 1993, where the shift supervisor had identified an expired working copy of procedure OP-126 (Main Steam, Extraction Steam, and Steam Dump Systems) during a tour of the steam tunnel area. Selected controlled copies of procedures are maintained at local operating stations to assist operators in system/component operation. The expired procedure was subsequently replaced with a current controlled copy of the procedure.

The inspector toured plant areas and verified that the procedures at local operating stations were properly controlled. Areas inspected included the steam tunnel, emergency diesel generator building, fuel handling building, and auxiliary control room. No discrepancies were identified. Also, the licensee's periodic audits of controlled procedures in these areas were reviewed. The audit reports indicated that these procedures were generally well maintained. No violations or deviations were identified.

b. Facility Tours and Observations

Throughout the inspection period, facility tours were conducted to observe operations, surveillance, and maintenance activities in progress. Some of these observations were conducted during backshifts. Also, during this inspection period, licensee meetings were attended by the inspectors to observe planning and management activities. The facility tours and observations encompassed the following areas: security perimeter fence; control room; emergency diesel generator building; reactor auxiliary building; waste processing building; turbine building; fuel handling building; emergency service water building; battery rooms; electrical switchgear rooms; the technical support center; and the emergency operations facility.

During these tours, the following observations were made:

(1) Monitoring Instrumentation - Equipment operating status, area atmospheric and liquid radiation monitors, electrical system lineup, reactor operating parameters, and auxiliary equipment operating parameters were observed to verify that indicated parameters were in accordance with the TS for the current operational mode.

(2) Shift Staffing - The inspectors verified that operating shift staffing was in accordance with TS requirements and that control room operations were being conducted in an orderly and professional manner. In addition, the inspector observed shift turnovers on various occasions to verify the continuity of plant status, operational problems, and other pertinent plant information during these turnovers.

(3) Plant Housekeeping Conditions - Storage of material and components, and cleanliness conditions of various areas throughout the facility were observed to determine whether safety and/or fire hazards existed.

(4) Radiological Protection Program - Radiation protection control activities were observed routinely to verify that these activities were in conformance with the facility policies and procedures, and in compliance with regulatory requirements. The inspectors also reviewed selected radiation work permits to verify that controls were adequate.

(5) Security Control - The performance of various shifts of the security force was observed in the conduct of daily activities which included: protected and vital area access controls; searching of personnel, packages, and vehicles; badge issuance and retrieval; escorting of visitors; patrols; and compensatory posts. In addition, the inspector observed the operational status of closed circuit television monitors, the intrusion detection system in the central and secondary alarm stations, protected area lighting, protected and vital area barrier integrity, and the security organization interface with operations and maintenance.

(6) Fire Protection - Fire protection activities, staffing and equipment were observed to verify that fire brigade staffing was appropriate and that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable.

(7) Between January and June 1993, significant amounts of scaffolding had been erected in the 190 foot elevation of the reactor auxiliary building in the vicinity of the "A" residual heat removal (RHR) and containment spray pumps, and above safety-related system piping, electrical junction boxes, conduits, and the area HVAC unit. The scaffolding had been erected to support the correction of problems which have occurred due to ground water intrusion (See NRC Inspection Report 50-400/92-04). Concurrently on May 19, 1993, the licensee performed a maintenance outage to repair several small deficiencies in "B" safety train components including the "C" CSIP, "B" RHR, CCW, and ESW pumps. The inspector questioned the wisdom of placing the "B" RHR

system under an equipment clearance which rendered it inoperable at the same time that scaffolding was erected around the "A" RHR pump and system piping. The inspector discussed this situation with licensee management. Although the licensee had established a program for the control of scaffolding, no seismic evaluations for erected scaffolding were required by the implementing procedure.

The inspector reviewed PLP-401 (Ladder, Scaffold, and Equipment Use and Storage) and noted specific exclusions for the erection of scaffolding over redundant safety trains simultaneously. The intent of this step was to prohibit the loss of both safety trains due solely to the failure of scaffolding. Regulatory Guide 1.29 discusses the "two-over-one" situation where nonsafety-related structures whose continued function is not required, but whose failure could reduce the functioning of any safety-related equipment, should be seismically designed and constructed. The inspector noted that procedure PLP-401 did not require a seismic evaluation of erected scaffolding over safety-related components; particularly when scaffolding is erected over the only available train of safety-related systems, as was the case on May 19.

The inspector related this concern to licensee management, who concurred with the inspectors comments and directed a seismic evaluation be performed for the "A" RHR and containment spray pump room. This evaluation concluded that the scaffolding would not damage safety-related components during a seismic event.

While the evaluation yielded favorable results for the scaffolding erected over the "A" RHR system, the inspectors concluded that the licensee's controls over this situation were weak, in that the evaluation should have been done prior to the removal of the "B" RHR system from service on May 19. The inspectors noted that the licensee has taken administrative measures to ensure that scaffolding is no longer erected over safety-related equipment until the procedure is enhanced to include better controls.

The inspectors found plant housekeeping and material condition of components to be satisfactory. The licensee's adherence to radiological controls, security controls, fire protection requirements, and TS requirements in these areas was satisfactory.

c. Review of Nonconformance Reports

Adverse Condition Reports (ACR) were reviewed to verify the following: TS were complied with, corrective actions and generic items were identified and items were reported as required by 10 CFR 50.73.

(1) ACR 93-220 reported that the electrical containment penetration for the Integrated Reactor Vessel Head (IRVH) bridge hoist was not provided with a secondary overcurrent protection device as specified in plant control wiring drawings. During a walkdown of these circuit breakers used to provide overcurrent protection on May 25, 1993, the licensee discovered that only one breaker provided power to the bridge hoist located inside containment. The other breaker specified on the drawings was found to be disconnected from the circuit. When notified of this condition, plant operators deenergized the circuit by establishing an equipment clearance in accordance with TS 3.8.4.1.a. The inspector verified the clearance tagout in the field. Although the tags were found to be on the specified components, the inspector noted that one of the breakers which were tagged (1D12-7BR) was labeled as a "spare" instead of a power source for the bridge hoist. This spare breaker coincided with the one which was not terminated to the circuit. The licensee's investigation into this matter concluded that this wiring deficiency had existed since initial plant construction.

The licensee has experienced previous problems with these types of electrical penetrations. On March 14, 1990, licensee personnel discovered a discrepancy in the amperage rating for a circuit breaker providing overcurrent protection. The licensee issued LER 90-08 reporting this condition. The corrective actions associated with this LER included a review of the calculations of similar circuits and a field verification that breakers used as protection devices were properly identified/verified. These actions were completed in May 1991. This matter was considered to be a licensee identified violation in NRC Inspection Report 50-400/90-06 (NCV 400/90-06-01).

Although the condition reported by ACR 93-220 was identified by the licensee, corrective actions performed for the previous deficiency should have identified the labeling problem for breaker 1D12-7BR and the inability of this secondary breaker to perform its function. Therefore, this matter is considered to be a violation of TS 3.8.4.a.

Violation (400/93-12-01): Failure to maintain the IRVH bridge hoist electrical containment penetration conductor overcurrent protective device operable.

(2) ACR 93-205 reported that on May 17, 1993, the RAB normal exhaust fan E-18 could not be secured from the control switch in the main control room. Auxiliary operators noted locally that the fan's circuit breaker was still closed and an electrical burning smell was present. The breaker subsequently automatically tripped open after two

unsuccessful manual attempts. Troubleshooting of this Asea Brown Boveri LK-16 circuit breaker revealed that the trip coil had burned up and the breaker had mechanically bound in the closed position. Although this application of the LK-16 breaker was in nonsafety-related equipment, many safety-related components receive electrical power through LK-16 breakers.

Several previous failures of these types of breakers have occurred in the past. As reported in LER 92-09, a failure of the E-18 supply breaker contributed to a reactor trip on July 15, 1992. As mentioned in NRC Inspection Reports 50-400/92-26, 50-400/92-13, and 50-400/90-13, the licensee has performed numerous corrective actions to improve breaker reliability including breaker physical modifications (PCR-3510, 480 volt Drawout Breakers used as Contacters) and periodic preventive maintenance checks. Although this action reduced the number of breaker failures, additional failures have occurred even on the modified breakers (the E-18 breaker had been previously modified in November 1990). The licensee plans to replace this switchgear beginning in the next refueling outage scheduled for March 1994 (PCR-6526, Frequently Cycled LK Breakers; PCR-6714, Frequently Cycled Non-Safety LK Breakers; PCR-6715, Train "B" Load Centers LK Breakers Replacement; and PCR-6896, Non-Safety Train "B" and General Services Bus Section 1 Breaker Replacement). Until then the licensee developed interim corrective action. The licensee determined that a significant number of the failures were from a few heavily cycled breakers. These breakers are being identified and "retired". In addition, the preventive maintenance frequency for the heavily cycled breakers will be increased to ensure breaker reliability.

On May 30, 1993, while attempting to deenergize nonsafety-related MCC 1-4B1 to perform LK-16 circuit breaker modifications per PCR-3510, plant operators could not open the 480 volt feeder breaker for this bus. Alternatively, the operators opened the corresponding 6.9 KV supply breaker to the MCC. Troubleshooting of this Asea Brown Boveri LK-42 feeder breaker revealed similar problems to those found on the LK-16 circuit breakers. The breakers are very similar in design. The licensee had already increased the scope of modification PCR-3510 to include all the LK breakers (i.e., LK-8, LK-16, LK-25, LK-32 and LK-42). Approximately 200 of the 243 LK circuit breakers have been modified. The modification was subsequently performed on this feeder breaker. The failure of this circuit breaker differed from previous failures in that the feeder breaker did not experience many cycles and was not anticipated to exhibit the same problems as the heavily cycled breakers. As discussed in paragraph 6, another failure mechanism for LK

breakers involves the closing coils. The inspectors will continue to monitor the licensee's progress in correcting these breaker problems.

(3) ACR 93-237 reported that the lock for the circuit breaker for valve 1CT-95, Containment Spray Pump 1B-SB Recirculation Valve, was not installed as required by Operations Management Manual procedure OMM-011, Control of Locked Valves. Valve 1CT-95 is a normally closed motor-operated containment isolation valve whose breaker is required to be locked/off during normal plant operating conditions, as described in the plant FSAR. During performance of surveillance procedure OST-1119, Containment Spray Operability Train B Quarterly Interval, on June 2, 1993, operators were required to remove the lock and close the breaker. This would allow for the manipulation of valve 1CT-95 to support surveillance testing of the "B" containment spray pump. Section 7.3, step 48 of the procedure requires operators to turn off and lock the breaker to valve 1CT-95 following the pump test. There are two sign-off steps in the procedure for this action, one for the operator performing the step and another for an independent verifier. Interviews with the involved personnel revealed that while performing the step, the operator turned the breaker off but failed to re-install the lock as required. Additionally, both sign-off steps were initialed by the operator and the independent verifier indicating that the lock had been installed and verified to be in place. The individuals involved indicated that the error was due to oversight and that they simply did not read the procedure step completely. The inspectors concluded that this was not a case of record falsification.

While this incident was identified by the licensee, it is the latest of several examples of plant performance where either self-checking or independent verification were lacking. On April 28, 1993, operations personnel went to the wrong breaker cubicle during efforts to swap safety trains. This resulted in an inadvertent start of the "B" CCW pump. In the past, NRC violations have been issued (Violations 400/91-27-01 and 400/91-09-01) for incidents involving inadequate self-checking or independent verification. One involved a mispositioned NI comparator switch which rendered the QPTR alarm inoperable in January 1992. The other involved instrument process tubing for a differential pressure switch in the EDG fuel oil system, which was neither reconnected nor properly verified to be connected prior to returning the instrument to service in April 1991. This latest example involving the failure to properly implement procedure OST-1119 is contrary to the requirements of TS 6.8.1.a and is being cited because violations have been previously identified by both the NRC

and the licensee in the areas of procedural adherence and independent verification, indicating a need for additional management attention in those areas.

Violation (400/93-12-02): Failure to properly implement plant procedures as required by TS 6.8.1.a.

3. Surveillance Observation (61726)

Surveillance tests were observed to verify that approved procedures were being used; qualified personnel were conducting the tests; tests were adequate to verify equipment operability; calibrated equipment was utilized; and TS requirements were followed. The following tests were observed and/or data reviewed:

- OST-1007 CVCS/SI System Operability Quarterly Interval
- OST-1039 Calculation of Quadrant Power Tilt Ratio, Weekly Interval (With Alarm Operable)
- OST-1079 Containment Isolation Valves ISI Test Quarterly Interval
- OST-1081 Containment Visual Inspection When Containment Integrity is Required
- MST-I0137 Main Steam/Feedwater Flow Loop 2 (F-0485/F-0486) Operational Test
- MST-I0138 Main Steam/Feedwater Flow Loop 3 (F-0494/F-0497) Operational Test
- MST-I0151 Steam Generator C Narrow Range Level Loop (L-0496) Operational Test
- EST-222 Procedure for the Type B LLRT of the Personnel Air Lock Barrel
- EPT-126T Turbine Volumetric Test

The performance of these procedures was found to be satisfactory with proper use of calibrated test equipment, necessary communications established, notification/authorization of control room personnel, and knowledgeable personnel having performed the tasks. The inspectors noted good use of the plant simulator for performing practice runs on procedure EPT-126T. This allowed operators to smoothly execute the actual test, which was an infrequently used procedure that had the potential for introducing unwanted plant transients if not performed correctly. No violations or deviations were observed.

4. Maintenance Observation (62703)

The inspector observed/reviewed maintenance activities to verify that correct equipment clearances were in effect; work requests and fire prevention work permits were issued and TS requirements were being followed. Maintenance was observed and work packages were reviewed for the following maintenance activities:

- Preventive maintenance on the startup transformer feeder breaker to auxiliary bus 1E (breaker 121) in accordance with procedure PM-E006, 6.9 KV 3000 Amp Air Circuit Breaker PM.

- Repair nitrogen leak on the actuator for the "A" main feedwater isolation valve 1FW-159 in accordance with procedure CM-M0059, Feedwater Isolation Valve Actuator Disassembly and Maintenance, Pilot Check Valve Shim Replacement and Fill/Bleed Procedure.

- Replacement of bent yoke that actuates circuit breaker 122 mechanism operated cell switch.

- Troubleshooting/calibration of the nitrogen pressure switch for the "C" main feedwater isolation valve 1FW-217 in accordance with procedure PIC-I100, Pressure and Differential Pressure Switch Inspection and Calibration.

- Repair of the "A" steam generator blowdown flow control valve 1BD-8 in accordance with procedure CM-M0033, ITT Hamel Dahl V500 Series Cage Trim Valves Disassembly and Maintenance.

- Troubleshooting the failure of the "B" chilled water pump breaker to close in accordance with procedures EPT-033, Emergency Safeguards Sequencer System Test, and PM-E0012, 480 VAC Load Center Breaker and Cubicle PM.

The performance of work was satisfactory with proper documentation of removed components and independent verification of the reinstallation. No violations or deviations were identified.

5. Safety Systems Walkdown (71710)

The inspector conducted a walkdown of portions of the fire protection system to verify that the lineup was in accordance with license requirements for system operability and that the system drawing and procedure correctly reflected "as-built" plant conditions.

The inspector noted that the material condition of components in the valve pit area adjacent to the motor driven fire pump was poor. The corresponding area near the diesel engine driven fire pump was in a much better material condition. The inspector discussed this observation with licensee personnel and discovered that efforts were already underway to upgrade the material condition of the motor driven fire pump components as had been recently done to the engine driven fire pump components.

From a review of plant drawings the inspector noted that no drawing existed for the fuel oil supply tank and delivery system to the diesel engine driven fire pump. Although only a few valves existed in this system (one flow isolation valve, one tank drain valve, and two level indicator isolation valves), the inspector considered the lack of a flow

drawing to be unusual. Licensee personnel stated that they would review this matter to determine if a drawing was appropriate.

No violations or deviations were identified.

6. Followup of Onsite Events (93702)

At 5:32 p.m., on May 23, 1993, the 1B-SB emergency bus was inadvertently deenergized which caused an automatic start of the "B" emergency diesel generator. A power increase was in progress following the outage. When operators attempted to transfer AC electrical power from the Startup Transformers (SUT) to the Unit Auxiliary Transformers (UAT) a circuit breaker malfunctioned which caused auxiliary bus 1E to be supplied from both the UAT and SUT. Operators proceeded to manually open the SUT feeder breaker (121) to restore the electrical lineup to the desired configuration. The auxiliary contacts associated with the UAT feeder breaker (122) were not properly engaged due to a damaged actuating mechanism and provided a false breaker open signal to the 1B-SB emergency bus feeder breaker (125) even though breaker 122 was actually closed. When breaker 121 was manually opened, feeder breaker 125 was provided with open signals from both breakers 121 and 122 and automatically opened as designed. This action deenergized the 1B-SB emergency bus and the emergency diesel generator started on a low bus voltage signal. Maintenance was performed on breaker 122 to replace the actuating mechanism for the auxiliary contacts which were found to be bent. The licensee believes the damage to the actuating mechanism occurred when the breaker was previously racked in during the October 1992 refueling outage. At 7:54 a.m., on May 24, the emergency diesel generator was secured and the electrical distribution lineup returned to the desired configuration with the UAT supplying auxiliary bus 1E.

During this event, the auxiliary feedwater system actuated as designed and a containment ventilation isolation signal was generated due to losing power to the containment system radiation area monitors. With the exception of the "B" chilled water pump which failed to start, plant equipment operated properly. The licensee investigated the problem and determined that the closing coil on the LK-16 circuit breaker for this pump had failed. The closing coil was subsequently replaced and the pump was tested satisfactory.

The licensee has experienced several previous problems with proper engagement of 6.9 KV breaker secondary contacts. As discussed in NRC Inspection Report 50-400/92-04, closing problems have been identified for these types of breakers. On October 1, 1988, a similar event to this one occurred. While in a refueling outage, power was inadvertently interrupted to the 1A-SA emergency bus during an attempt to realign the electrical distribution system to backfeed electrical power through the main transformers. In this event the SUT feeder breaker to auxiliary bus 1D failed to open when the UAT feeder breaker was closed. When operators manually opened the SUT feeder breaker the 1A-SA feeder breaker automatically opened and deenergized the emergency bus. Troubleshooting for this condition also revealed that the actuating

mechanism for the auxiliary contacts on the UAT feeder breaker were bent. Corrective action to prevent recurrence of this event included a repair to the actuating mechanism, and procedural guidance provided to operators to prevent actuating mechanism damage when racking 6.9 KV breakers into the connected position. Operating procedure OP-156.02, AC Electrical Distribution, was revised to provide a note alerting operators to visually ensure that the cell switch is properly aligned with the actuating mechanism.

The failure to properly implement procedure OP-156.02 (i.e., verify the alignment of the circuit breaker actuating mechanism with the cell switch when the breaker was placed into service during the previous refueling outage) is contrary to the requirements of TS 6.8.1.a and is considered to be another example of the violation discussed in paragraph 2.c.(3) of this report.

7. Outage Activities (71707)

Major work performed during this planned outage included repairs to steam generator blowdown valve 1BD-8, oil addition to the "B" reactor coolant pump, repair of nitrogen leaks on the operators for valves 1FW-159 and 1FW-217, replacement of the motor for the "A" heater drain pump, and repair of through-wall leaks on the seal water supply piping for the "B" condensate booster pump.

As mentioned in NRC Inspection Report 50-400/93-08, containment sump inleakage had increased and a temporary modification had been installed to valve 1BD-8 to backseat the valve and reduce secondary system inleakage to the sump. During this inspection period, a gradual increase in sump inleakage occurred. Licensee management decided to take the unit off line and repair this valve prior to the upcoming summer peak load period.

Significant planning activities included the use of a spare "mockup" valve and actuator for the 1BD-8 work and a spare reactor coolant pump motor for the oil addition. The simulator was utilized to provide operators with unit shutdown and startup training, as well as potential transients which could be expected during these plant evolutions (i.e., feedwater transient and premature reactor criticality). The inspector found this pre-planning to be effective and the actual evolutions were performed without mishap.

Additional activities performed by the licensee during the outage included a containment visual inspection to identify fibrous air filters or other sources of fibrous material in accordance with NRC Bulletin 93-02, Debris Plugging of Emergency Core Cooling Suction Strainers, and a visual inspection of the reactor coolant pressure boundary for boric acid leakage. No fibrous material and only minor boron residue was found. Work tickets were generated to correct the boric acid leakage during the next refueling outage.

The inspector requested the documented inspection results for the boric acid leakage inspection. The inspector was informed that the results were not well documented. Procedure PLP-600, Boron Corrosion Program, and inspection procedure OPT-1519, Containment Visual Inspection for Boron Leakage Every Outage Shutdown, specified that a walkdown inspection be performed prior to cooldown in Mode 3. Since a plant cooldown was not commenced, the inspection procedure, which included an attachment specifying areas to be inspected and results, was not implemented. After the inspector discussed this matter with licensee management, the data sheets were filled out with the results for the areas inspected. The inspector considered the licensee's documentation of the boric acid leak inspection to be poor.

8. Adverse Weather Operations (71707)

On June 4, 1993, a tornado warning was issued for Wake County. The inspector reviewed the licensee's preparations performed in response to this warning. No tornadoes approached the plant and no plant damage occurred during the adverse weather conditions. The inspector reviewed the provisions contained in the licensee's emergency plan for handling these situations and also reviewed procedure AP-301, Adverse Weather Operations. The licensee's emergency plan requires the declaration of an Unusual Event if a tornado crosses the exclusion area boundary, and if sustained wind speeds exceed 90 MPH then an ALERT would be declared. If a tornado impacts the power block and sustained winds exceed 100 MPH then a site emergency would be declared. Procedure AP-301 contains provisions to ensure the plant is placed in hot standby (Mode 3) at least two hours prior to the anticipated arrival of sustained winds in excess of 73 MPH. For the tornado warning issued on June 4, loose material was removed from exposed areas, access doors closed, safety lines fixed, and emergency equipment was verified to be available. The inspector found that the licensee's emergency plan and procedures were properly implemented.

9. Review of Licensee Event Reports (92700)

The following LERs were reviewed for potential generic impact, to detect trends, and to determine whether corrective actions appeared appropriate. Events that were reported immediately were reviewed as they occurred to determine if the TS were satisfied. LERs were reviewed in accordance with the current NRC Enforcement Policy.

a. (Open) LER 93-05: This LER reported an entry into TS 3.0.3 on April 28, 1993 during an in-service stroke test on two CSIP discharge cross connect valves. As discussed in NRC Inspection Reports 50-400/93-08 and 50-400/93-10, the "B" CSIP rotating assembly was replaced following a shaft failure. Following the return to service for this pump, valves 1CS-218 and 1CS-220 were stroke tested to satisfy IST requirements. These valves isolate the "A" CSIP from the normal high head safety injection flow path when closed. The valves' stroke times were approximately ten seconds each and they were closed for a total of less than one

minute. On April 29, it was determined that the "B" CSIP had not yet been demonstrated fully operable and that both ECCS flow paths were inoperable for the seconds that the valves were closed. This constituted a TS 3.0.3 entry. The "C" CSIP was subsequently placed into service and the "B" CSIP was declared inoperable.

The licensee attributed this event to several causes, all stemming from personnel error. Inadequate evaluation and review of the "B" CSIP test data coupled with a lack of understanding on the use and restrictions of the JCO process were cited as reasons for the premature declaration of the pump to be operable. The licensee will train appropriate plant personnel on this event and the proper use of JCOs, Engineering Evaluations, Technical Specifications, and other processes applicable to this event. Additionally, the "B" CSIP will now remain inoperable until conditions permit full flow testing to verify the pump's ability to meet vendor and TS runout limits. This LER will remain open pending completion of the above actions.

b. (Open) LER 93-06: This LER reported two separate incidents where the requirements of TS 4.11.1.1.1 were not met. Specifically, on May 17 and May 31, 1993, it was determined that the automatic sampling device on the Secondary Waste Sample Tank (SWST) effluent line had malfunctioned during releases from the SWST. Technical Specification 4.11.1.1.1 requires continuous sampling of the SWST effluent during releases to provide a weekly radioactivity composite. The composite sample would be used as a backup to the normal radiation monitor in the event it became inoperable. This composite sample is typically discarded when the radiation monitor is operable. In both instances the SWST releases were immediately secured and the in-line radiation monitor was used to verify that no detectable levels of radioactivity existed.

The sampling device malfunctions were caused by a failure in the unit's electronic counter. After the second failure, the licensee consulted with the vendor who attributed the cause of both failures to inadequate surge protection in the counter power supply. Immediate corrective actions were taken to replace the automatic sampling device with a modified version which included enhanced surge protection for the counter power supply. Long-term corrective actions are to include developing test procedures to more thoroughly test or calibrate the composite sampling units. This LER will remain open pending completion of the procedure development.

10. Licensee Action on Previously Identified Inspection Findings (92702 & 92701)

(Closed) URI 400/93-10-01: Review calculations and licensee actions regarding the seismic qualification of AFW piping.

As discussed in NRC Inspection Report 50-400/93-10, the inspectors identified two valves in the AFW system, 1AF-5 and 1AF-24, which appeared to be installed without adequate supports. These two-inch motor-operated valves are in the recirculation lines from the two motor-driven AFW pumps to the condensate storage tank. The valve bodies are installed horizontally in the piping with big Limitorque motor operators attached to their sides. The motor operators had no additional supports other than the small two-inch piping, and seemed to impose a significant bending moment on that piping. The inspectors brought this observation to the licensee's attention who researched the original design calculations for that section of piping and discovered a modeling error in the centers of gravity for the two valves. Specifically, Stress Analysis Calculation 71-1, "AF Piping From Mass Points 66 and 453 at Floor (Elevation 261') to Steam Generator Auxiliary Feedwater Pumps Discharge Nozzles and Anchor Point 4903 at Wall", incorrectly applied the mass of the composite valve body/motor operator assemblies at the center of gravity of the valve bodies themselves for valves 1AF-5 and 1AF-24, and not at the center of gravity of the composite valve assemblies. Stress calculations resulting from this modeling error incorrectly yielded stress figures which were within the design basis allowable stress criteria as referenced in the FSAR, Table 3.9.3-11.

Recalculations of the actual pipe stresses imposed by the unsupported motor operators yielded stress values of approximately 33 ksi. This figure exceeded the allowable value for the emergency condition (which, as defined by FSAR Table 3.9.3-7, included the safe shutdown earthquake in its loading combination) by 6 ksi. This rendered that section of the seismic class I AFW system outside of its original design allowable stress limits. Upon discovering this, the licensee performed an immediate operability determination as allowed by the guidelines of NRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability" and the licensee's own Design Guide No. DG-II.20, "Design Guide for Civil/Structural Operability Reviews". The design guide incorporated more recent rules for calculating stress limits as specified in Appendix F of the Appendices to ASME Boiler & Pressure Vessel Code, Section III, 1986 edition. Specifically, a stress limit of two times the material yield stress of 36 ksi, or 72 ksi, could be used in determining the short-term structural integrity of the AFW recirculation piping. Based on the actual stresses being calculated at 33 ksi, the system was considered operable in the short-term.

Stress Analysis Calculation 71-1, which contained the center of gravity modeling error was originally developed in 1974 by the plant's architect engineer. The calculation received reviews by the licensee in 1986 in accordance with the guidelines established in NRC Bulletin 79-14, "Seismic Analyses for As-Built Safety-Related Piping Systems". However, those reviews failed to identify the modeling error which resulted in that section of AFW piping being outside of its design basis. The failure to perform adequate measures to ensure the accuracy of design with respect to this seismic class I system is considered to be a violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control.

Violation (400/93-12-03): Failure to establish adequate measures to verify that designs were technically accurate with respect to the design basis for the AFW system.

In accordance with the requirements of NRC Generic Letter 91-18 and the licensee's design guide, an engineering evaluation which will document the current, short-term operable condition, as well as a modification which will restore the system to long-term acceptable status, are being developed by the licensee.

11. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on June 21, 1993. During this meeting, the inspectors summarized the scope and findings of the inspection as they are detailed in this report, with particular emphasis on the Violations addressed below. The licensee representatives acknowledged the inspector's comments and did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. No dissenting comments from the licensee were received.

Item Number	Description and Reference
400/93-12-01	VIO: Failure to maintain the IRVH bridge hoist electrical containment penetration conductor overcurrent protective device operable, paragraph 2.c.(1).
400/93-12-02	VIO: Failure to properly implement plant procedures as required by TS 6.8.1.a., paragraphs 2.c.(3) and 6.
400/93-12-03	VIO: Failure to establish adequate measures to verify that designs were technically accurate with respect to the design basis for the AFW system, paragraph 10.

12. Acronyms and Initialisms

ACR - Adverse Condition Report
AFW - Auxiliary Feedwater
AP - Administrative Procedure
ASME - American Society of Mechanical Engineers
CCW - Component Cooling Water
CFR - Code of Federal Regulations
CSIP - Charging Safety Injection Pump
CVCS - Chemical Volume Control System
ECCS - Emergency Core Cooling System
EDG - Emergency Diesel Generator
EPT - Engineering Performance Test

EST	-	Engineering Surveillance Test
ESW	-	Emergency Service Water
FSAR	-	Final Safety Analysis Report
HVAC	-	Heating, Ventilation and Air Conditioning
IRVH	-	Integrated Reactor Vessel Head
ISI	-	Inservice Inspection
IST	-	Inservice Testing
JCO	-	Justification for Continued Operation
KV	-	Kilovolt
KSI	-	Kilopounds per square inch
LER	-	Licensee Event Report
LLRT	-	Local Leak Rate Test
MCC	-	Motor Control Center
MPH	-	Miles Per Hour
NCV	-	Non-Cited Violation
NRC	-	Nuclear Regulatory Commission
OPT	-	Operations Performance Test
OST	-	Operations Surveillance Test
PCR	-	Plant Change Request
PLP	-	Plant Procedure
PM	-	Preventive Maintenance
QPTR	-	Quadrant Power Tilt Ratio
RAB	-	Reactor Auxiliary Building
RHR	-	Residual Heat Removal
SI	-	Safety Injection
SUT	-	Startup Transformers
SWST	-	Secondary Waste Sample Tank
TS	-	Technical Specification
UAT	-	Unit Auxiliary Feedwater
URI	-	Unresolved Item
VAC	-	Volt Alternating Current
VIO	-	Violation