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

Report No.: 50-400/95-15

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

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

Facility Name: Harris 1

Inspection Conducted: September 3 - October 7, 1995

Inspector: D. Wisenor for D. Roberts, Acting Senior Resident Inspector 11/3/95 Date Signed

Other Inspector: S. Elrod, Senior Resident Inspector

Approved by: M B Shymlock M. Shymlock, Chief 11-3-95 Date Signed
Reactor Projects Branch 4
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted in the areas of operations, maintenance, surveillance, engineering, plant support, review of licensee event reports, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed.

Results:

Plant Operations

Midloop operations, refueling activities, and control room response to a smoking breaker cubicle were good. Contractor performance during core manipulations was good. Personnel and process errors were noted in other areas which warranted increased licensee management attention, paragraph 3.b. An inadequate procedure resulted in an unplanned actuation of the turbine driven auxiliary feedwater pump and an LER, paragraph 3.f.(1). The inspectors identified one violation in the Operations area which was significant because it involved a situation where operators were not aware of the inoperable status of the "B" CSIP, causing them to place it in service on September 2 without it being properly tested, paragraph 3.a.(1).

Maintenance

Overall, maintenance and surveillance activities were conducted well during the outage. However, one violation was caused by an inadequate procedure which led to an unplanned partial safety injection, paragraph 5.c.

Engineering

Overall, engineering activities were performed well, especially considering the many engineering challenges that were presented during the refueling outage. The inspectors identified no violations or deviations in the engineering area.

Plant Support

Licensee performance in the radiological controls area was good during the refueling outage, paragraph 6.b. PNSC meetings had good safety focus, paragraph 6.f.

REPORT DETAILS

1. PERSONS CONTACTED

Licensee Employees
D. Batton, Superintendent, On-Line Scheduling
D. Braund, Manager, Security
J. Collins, Manager, Training
*J. Dobbs, Manager, Outage and Scheduling
*J. Donahue, General Manager, Harris Plant
R. Duncan, Superintendent, Mechanical Systems
*W. Gautier, Manager, Maintenance
M. Hamby, Manager, Regulatory Compliance
*M. Hill, Manager, Nuclear Assessment
D. McCarthy, Superintendent, Outage Management
*R. Prunty, Manager, Licensing and Regulatory Programs
*W. Robinson, Vice President, Harris Plant
*G. Rolfson, Manager, Harris Engineering Support Services
S. Sewell, Superintendent, Design Control
*T. Walt, Manager, Regulatory Affairs
*B. White, Manager, Environmental and Radiation Control
*A. Williams, Manager, Operations

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

NRC Personnel
C. Bajwa, Systems Engineer, Office of Nuclear Reactor Regulation (NRR)
*S. Elrod, Senior Resident Inspector, Harris Plant
*C. Lui, Risk Assessment Engineer, Office of Nuclear Regulatory Research
*D. Roberts, Resident Inspector, Harris Plant
L. Whitney, Project Manager, NRR
F. Wright, Senior Radiation Specialist, Region II

*Attended exit interview

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

2. PLANT STATUS AND ACTIVITIES

a. Operating Status of the Plant Over the Inspection Period.

The plant began the inspection period being cooled down in preparation for refueling. At the time, the plant was in Hot Shutdown (Mode 4) with RCS temperature and pressure about 325 °F and 350 psig, respectively. Operators continued to cool and depressurize the plant, entering Cold Shutdown (Mode 5) on September 3. The plant entered Refueling (Mode 6) on September 9. Fuel was completely offloaded from the reactor vessel from September 12 to 15. The plant re-entered Mode 6 on September 26 when the first fuel bundle was reloaded into the

reactor vessel. Core reload was complete on September 28. The plant entered Mode 5 on October 2 when the reactor vessel head studs were fully tensioned. The plant commenced the post-outage heatup on October 7, entering Mode 4 that morning and Mode 3 that afternoon. The plant ended the inspection period in Mode 3 with RCS heatup in progress and RCS temperature and pressure approaching normal operating conditions.

b. Other NRC Inspections or Meetings at the Site.

F. Wright, Senior Radiation Specialist, NRC RII, was on site from September 18-22 conducting an inspection in the area of radiological control and protection. The inspector conducted an exit meeting on September 22 and his findings were documented in IR 400/95-14.

C. Lui, Risk Assessment Engineer, NRC Office of Nuclear Regulatory Research, was on site from September 11 - October 6 observing outage activities.

C. Bajwa and L. Whitney, both of the NRC Office of Nuclear Reactor Regulation, were on site on October 5 and 6 viewing plant fire protection modifications and studying post-fire safe shutdown procedures. The NRC representatives were accompanied by Messrs. T. Storey and K. Sullivan, both NRC contractors, and Mr. N. Berkoff, a U.S. Department of State contract interpreter. Also accompanying the NRC staffmembers were fourteen foreign visitors representing the Russian Federation regulatory body, Russian industrial fire protection government ministry, Russian power reactor plant operating organization, Ukrainian regulatory body, Ukrainian industrial fire protection/fire fighting and research organizations, and Czech Republic and Hungarian regulatory bodies.

3. OPERATIONS

a. Plant Operations (71707)

(1) Shift Logs and Facility Records

The inspector reviewed records and discussed various entries with operations personnel to verify compliance with the TS and the licensee's administrative procedures. In addition, the inspector independently verified clearance order tagouts.

<u>Operation in Mode 4 with Potentially No Operable CSIPs</u>

With the plant in Mode 4 on September 2, the inspector reviewed the Shift Supervisor's logbook and discovered that operators had declared operable and placed in service the "B" charging/safety injection pump (CSIP) without properly testing it. The "A" and "C" CSIPs were removed from service

and declared inoperable to satisfy LTOP requirements contained in TS 3.1.2.1 (Boration Systems Flowpath) and TS 3.5.3 (ECCS Subsystems) with RCS temperature less than 325 °F. The "C" pump's electrical breaker was racked out and the "A" pump's manual discharge valve had been locked closed to prevent either pump from injecting with the plant in a pressurized, low temperature condition. Prior to this configuration and for much of the previous operating cycle, the "A" and "C" pumps had been in service satisfying Mode 1-3 TS requirements. The following paragraphs discuss how the "B" pump should not have been declared operable on September 2 since it had not been properly tested prior to placing it in service.

Background

Each CSIP has a miniflow system that ensures a vendor-recommended minimum flowrate of 60 GPM for pump protection.

The "B" CSIP had been out of service and inoperable for much of the previous operating cycle due to a failed check valve (1CS-193) in its miniflow system. A stuck disk caused the valve to experience significant backflow leakage over the cycle requiring that the "B" pump not be placed in service concurrent with the "A" pump. The backflow leakage did not affect operability of the "B" pump, which only required that the check valve pass forward flow. However, a postulated "B" train electrical failure would take out the "B" pump, and the undesirable check valve leak path could potentially divert enough "A" pump flow during a LOCA, thus preventing the "A" pump from performing its safety function.

Because of the backflow leakage problems, in late July licensee personnel replaced the check valve (a 2-inch T-style globe check valve with a resilient seating surface) with another T-style valve minus the soft seat. Later, on August 2, the inspector observed the newly installed check valve fail a forward flow test, passing approximately 28 gpm, vice 60 gpm. The failure was caused by a sticking disk inside the valve. After this failure, between August 2 and September 2, operators maintained compliance with Mode 1-3 Technical Specifications by using the "C" pump to replace the "B" pump and isolating the "B" pump's miniflow leak path from the "A" pump.

The inspector, recalling that successful completion of the forward flow test was a prerequisite for "B" pump operability, questioned its operability on September 2. This information was indicated to the control room. The control room operators were unaware of the earlier 1CS-193 forward flow failure. After researching the previous month's test records and the reactor operator's logbook

entry for August 2, the operators determined that the "B" pump had indeed not been successfully tested prior to placing it in service on September 2. At the time of this discovery, RCS temperature was less than 325 °F. Operators immediately corrected the situation by raising RCS temperature above 325 °F to ensure compliance with LTOP requirements, opening the "A" pump's discharge isolation valve, and placing that pump in service. After securing the "B" pump and racking out its breaker, operators continued the plant cooldown to Mode 5.

Root Cause

Operators improperly placed the "B" pump in service because they were unaware of its inoperable status since August 2. Further review determined the root cause to be inadequate Equipment Inoperable Records (EIRs) for both valve 1CS-193 and the "B" CSIP. The EIRs did not reference the valve's forward flow test failure. EIRs were used to document inoperable equipment, failed surveillance tests, and associated retest requirements. These forms were relied upon by operators to ensure that TS LCOs were complied with when manipulating or testing plant equipment.

The EIRs associated with the "B" CSIP and valve 1CS-193 were initiated in July because of the previous backflow test failure, but were not updated by the SCO on August 2 when the newly replaced valve failed the forward flow test. Responsible plant personnel incorrectly assumed on August 2 that because the valve was already inoperable prior to the forward flow test, the EIRs generated in July were properly annotated with retest requirements. Since "B" pump miniflow check valve backleakage only affected "A" pump operability, the "B" pump EIR only referenced that the pump's breaker had to be racked in to make it operable. Likewise, the only retest requirement referenced on the 1CS-193 EIR was its backflow leak test. Successful completion of this test would be required prior to unisolating the valve from the "A" pump to maintain that pump's operability.

On September 2, because of the incomplete information in the EIRs, operators placed the "B" CSIP in service to support an upcoming 18-month surveillance test without forward flow testing the miniflow check valve, and made the other pumps inoperable. The plant was operated in this condition for approximately seven hours before the inspector discovered it. During that time, operators were cooling down the RCS in accordance with operating procedures, thereby adding positive reactivity to the core with one degraded CSIP functional.

Operations Management Manual procedure OMM-014, Rev. 4, Operation of the Work Control Center, Step 5.3.6 required that the SCO annotate all applicable EIRs when equipment failed surveillance testing or portions of the test which were not completed. Procedure OMM-014 further required that the SCO do this by entering all pertinent information in the Remarks section of the EIR, which was Attachment 4 to the procedure. Procedure OMM-007, Rev. 4, Operations Surveillance, Periodic and Reliability Tests, required that the Shift Supervisor ensure that an EIR was completed when an OST failed to meet its acceptance criteria. The failure to complete the "B" CSIP and valve 1CS-193 EIRs with forward flow test information resulted in the plant operating in Mode 4 for over 7 hours with one degraded CSIP. The failure to follow procedures OMM-007 and OMM-014 was contrary to the requirements of TS 6.8.1a and Regulatory Guide 1.33, Rev. 2, Appendix A which required procedures for equipment control. This is identified as Violation 400/95-15-01, Failure to Properly Annotate Surveillance Test Requirement for an Inoperable CSIP.

Safety Significance

Although this violation resulted in the "B" CSIP being placed in service without testing its miniflow system, the safety consequences were minimal. At the time of the discovery, the inspector verified that at least a 60 gpm flowrate existed via the normal charging/seal injection flowpath. Additionally, the licensee initiated ESR 9500752 to show that the "B" pump never operated below its minimum flow limit and that it could have performed its safety function with the plant in Mode 4 and RCS pressure reduced to 350 psig. The licensee's evaluation is further discussed in paragraph 5.a.4 of this report. While the event's safety significance was minor, the problem could have resulted in a substantial safety hazard under different plant operating conditions (Modes 1, 2 or 3).

Corrective Actions

To correct the valve problem, the licensee replaced the T-style globe check valve with a Y-type valve. The new valve had a disk arrangement designed to operate better under higher system pressures. The valve subsequently passed both forward flow and backflow leakage tests. To correct the EIR documentation problem, operations management issued night orders describing the event and discussing the need for more attention to detail in documenting surveillance test results.

The licensee documented this event in LER 95-08 which is closed in paragraph 3.f.(2) of this report. The licensee also reported the check valve deficiency in accordance with 10 CFR 21.21.

The inspectors found the logs and other facility records to be legible and well organized, and to provide sufficient information on plant status and events. The inspectors found clearance tagouts to be properly implemented. The inspectors identified one violation in this area.

(2) Facility Tours and Observations

Throughout the inspection period, the inspectors toured the facility to observe activities in progress, and attended several licensee meetings to observe planning and management activities. Inspectors made some of these observations during backshifts.

During these tours, the inspectors observed monitoring instrumentation and equipment operation. The inspectors also verified that operating shift staffing met TS requirements and that the licensee was conducting control room operations in an orderly and professional manner. The inspectors additionally observed several shift turnovers to verify continuity of plant status, operational problems, and other pertinent plant information. Licensee performance in these areas was satisfactory.

The inspectors identified no violations or deviations in the facility tours and observations area.

b. Effectiveness of Licensee Control in Identifying, Resolving, and Preventing Problems (40500)

Condition Reports (CRs) were reviewed to verify that TS were complied with, corrective actions and generic items were identified, and items were reported as required by 10 CFR 50.73.

Several CRs documenting personnel or process errors were generated during the refueling outage. While the numerous CRs indicated that the licensee's threshold for documenting nonconformances was relatively low, some of these CRs described incidents that warranted additional management attention in the area of work performance and clearances. The following incidents documented in CRs were noteworthy:

- Operator error in conducting the Loss of Offsite Power/LOCA sequencing test resulted in the "A" RHR pump starting twice in 20 seconds, as discussed in paragraph 4.b.(2) of this report.

- Due to a process error, operators inadvertently placed SG PORVs "A" and "C" under clearance. Per the licensee's shutdown risk assessment, the valves were assumed to be part of a key safety function available during cold shutdown, maintaining the SGs as a diverse decay heat removal method. When operators realized the error, the "A" and "C" SG PORVs were returned to functional status. Fortunately, defense in depth was not compromised as the RHR pumps were both operable and in service at the time.

- Upon installing an inappropriate clearance, operators isolated instrument air to the containment building, thereby isolating the normal CVCS letdown flowpath. The clearance also caused a service water header discharge isolation valve to unexpectedly close, essentially removing one train of ESW from service. Fortunately, RCS letdown was aligned to the RHR system at the time which eliminated the potential for unexpected RCS heatup. Operators promptly restored the service water header by placing the normal service water system in service.

The above incidents, including the inoperable CSIP event discussed in section 3.a.(1), all occurred during the first few days of the outage. While errors like these were reduced as the outage continued, other performance deficiencies surfaced (improperly restored clearances, poor FME controls) which warranted increased management attention in this area.

c. Refueling Activities (71707)

The inspector observed fuel offload and reload activities in accordance with fuel handling procedures FHP-014, Rev.6, Fuel and Insert Shuffle Sequence, and FHP-020, Rev. 7/3, Refueling Operations. Fuel handling equipment, including the refueling bridge crane, hoist, and load cell had been properly tested, inspected and calibrated prior to fuel movement, as required by plant procedures. The fuel handling equipment performed well during the evolutions. Operators maintained the refueling cavity water level at 23 feet above the reactor vessel flange during fuel movement. The licensee's FME program was in effect, with areas around and above the refueling cavity properly controlled as FME boundaries. An FME coordinator signed people and equipment in and out of the FME zone as required by plant procedures. Potentially loose articles were properly tied off. The inspector observed that the fuel movement was directly supervised by a licensed senior reactor operator, as required by TS 6.2.2d. Licensee and contractor personnel performed these core manipulations in a skilled and professional manner.

d. Midloop Operations (71707)

Due to continuing work in the steam generators, the plant had to go into midloop operations after core reload during the weekend of September 30. The inspectors verified that midloop and reduced inventory activities were conducted in accordance with expectations contained in NRC Generic Letter 88-17, "Loss of Decay Heat Removal." Specifically, the inspectors verified that procedural controls were in place and certain systems were available throughout the operation. Additionally, the inspectors ascertained that operators were trained and pre-briefed on the evolution; and were aware of associated risks.

Operators received control room training on midloop operations just days prior to the evolution. Additionally, pre-job briefings were conducted for each involved shift. Procedures covering the evolution included General Procedure GP-08, Draining the Reactor Coolant System; and OMP-004, Control of Plant Activities During Reduced Inventory Conditions. These procedures required at least two independent core exit temperature indications, two independent reactor vessel water level indications, and two additional means of adding inventory to the RCS be available. Containment closure requirements were contained in OST-1091, Containment Closure Test, Weekly Interval; and OST-1034, Containment Penetrations Test, Weekly. The inspectors verified that all system availability and containment closure requirements were met prior to and during reduced inventory operations. Additionally, only two of the three steam generators had nozzle dams installed, so the reactor vessel was adequately vented.

The inspector concluded that the licensee's controls for midloop operations were adequate. Additionally, operators performed draindown activities well.

e. Plant Response to Fire Announcement (71707)

The inspectors observed plant response to a perceived switchgear fire. On September 29, during a containment spray actuation test procedure, an Asea Brown Boveri Model LK-16 breaker failed to trip open as required. This breaker, located in non-safety related switchgear cubicle 1D1-4C, served a containment pre-entry purge fan, E-5B, which was supposed to trip on low flow following the containment spray actuation. Personnel observed smoke coming from the breaker cubicle and communicated this to the main control room. After control room personnel sounded the plant fire alarm, the shift supervisor announced the location of the "fire" and requested fire brigade response. When the inspector arrived at the switchgear, there was no fire and personnel had racked out the breaker. There was a burning smell and personnel explained that there had been no fire, but the trip coil in the breaker had overheated and released smoke into the area. This had been a recurring problem over the years with 480 volt LK-16 breaker trip

coils. Over the last two refueling outages, the licensee replaced all LK-16 safety-related breakers and frequently cycled non-safety related breakers with a Siemens model. This breaker was not considered a frequently cycled breaker and was not replaced.

The plant fire brigade responded to the control room announcement in a timely manner. The inspector concluded that the control room's prompt response and cautious actions regarding the perceived "fire" was excellent.

f. Review of LERs (92700)

(1) (Open) LER 95-007-00, Inadvertent Start of the Turbine Driven AFW Pump/Unplanned ESF Actuation and Identification of an Additional Related Test Deficiency.

This LER discussed a procedural deficiency that, during an "A" train safety bus undervoltage relay logic test, caused an inadvertent start of the turbine driven AFW pump and resultant feeding of the steam generators. During the investigation of the pump actuation, the licensee discovered another reportable procedural deficiency related to testing of undervoltage relays. Both items were reported on LER 95-07 and are discussed below.

On September 1, plant personnel performed procedure MST-E0034, 6.9KV Emergency Bus, 1A-SA Under Voltage (Loss of Voltage) Channel Calibration. The plant was in Mode 1 at approximately 75% reactor power and the procedure was a post maintenance test following calibrations on each of three 6.9kV bus 1A-SA undervoltage relays. The procedure called for depressing a test push-button which caused actuation of both the undervoltage lockout relay 86UV and the test lockout relay 86T. The 86T test relay blocked signals from the 86UV relay and its associated relays except for a signal from relay 86UVX to the TDAFW pump steam supply valve, 1MS-70. Since that signal was not blocked, and the logic for opening the valve on a bus undervoltage was satisfied, the valve stroked open and provided steam to the Terry turbine.

During an investigation into the above event, licensee personnel discovered another TDAFW pump testing deficiency related to the 86UVX relay. Specifically, TS 4.3.2.1 and TS Table 4.3-2 contained monthly Trip Actuating Device Operational Test (TADOT) requirements applicable to both the motor driven AFW pumps and the TDAFW pump. The motor driven pump requirement was covered by procedure OST-1124, 6.9kV Emergency Bus Undervoltage Trip Actuating Device Operational Test Monthly Interval Modes 1-2-3-4, which had personnel visually verify that the 86UV relay rolled during testing. Plant procedure writers incorrectly assumed that the turbine driven pump was also covered by visual observation of the

86UV relay, and did not understand that the associated 86UVX relay was also in the starting circuit. Thus, the monthly test procedure did not contain directions to verify actuation of the 86UVX relay, and the monthly TS requirement was never covered.

Licensee personnel determined that the root cause of both events was procedural error. Related to the inadvertent pump actuation, plant surveillance procedures generally contained mode restrictions and statements cautioning personnel on plant conditions which may affect test performance. The bus undervoltage test procedure MST-E0034 allowed the test to be performed in Modes 1 through 6, inclusive, and contained no precautions describing what would happen if the test was performed in Modes 1 though 4 with steam available to the turbine. The test was normally performed in Modes 5 and 6 with no plant steam such that when valve 1MS-70 opened, the TDAFW pump would not start. Because of the procedural omissions, personnel performing the procedure on September 1 were unaware of the potential pump start.

The safety significance of both issues was minimal. There were no adverse affects on safe plant operation, no pump damage, and no resultant inoperability of safety systems following the inadvertent TDAFW pump start. Operators in the control room secured AFW flow to the steam generators in a timely manner. Although the 86UVX relay was never verified on a monthly basis as required by TS, its actuation on bus undervoltage was verified every 18 months by deenergizing the safety bus during the TS-required EDG operability tests.

The licensee's corrective actions will include revising the procedures to correct the surveillance test deficiencies.

This LER will remain open pending licensee completion and inspector review of corrective actions.

(2) (Closed) LER 95-008-00, "B" Charging/Safety Injection Pump was Returned to Service Prior to Required Acceptance Testing, Resulting in Technical Specification Violation.

This inspector identified violation was discussed in paragraph 3.a.(1) of this report. The LER is closed and the licensee's corrective actions will be tracked with Violation 400/95-15-01.

g. Followup - Operations (92901)

(Closed) IFI 95-11-01, Questionable Position Indication for Containment Isolation Valve 1SP-209.

The inspector consulted NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants, to determine the need for verifying both the open and closed positions for valves having one safety function. In this case, valve 1SP-209 gave dual (mid-position) indication in the control room on May 11 when operators took its control switch to the open position. The valve in question was a small sealed solenoid valve for which only remote position indication was available. The valve's safety function was to close on a containment isolation signal. Licensee management determined that, since the closed position was never in question, the valve remained operable from May 11 until the valve's next inservice test became due. At that time both the open and closed positions would need verification for stroke timing as required by ASME Section XI Inservice Testing Requirements, and implemented by the licensee's program procedure ISI-203, ASME Section XI Pump and Valve Program Plan.

From the review of NUREG-1482, the inspector determined that no additional verification was required for the valve to remain operable on or after May 11. Stroke time testing on June 12, which included timing the valve stroke from fully open to fully closed (based on control room indication) was adequate. Further valve operations did not give the same dual position indication.

This item is closed.

Midloop operations, refueling activities, and control room response to a smoking breaker cubicle were good. Contractor performance during core manipulations was good.. Personnel and process errors were noted in other areas which warranted increased licensee management attention. The inspectors identified one violation in the Operations area which was significant because it involved a situation where operators were not aware of the inoperable status of the "B" CSIP, causing them to place it in service on September 2 prior to it being properly tested.

4. MAINTENANCE

a. Maintenance Observation (62703)

The inspector observed the maintenance and reviewed the work packages for the following maintenance activities to verify that correct equipment clearances were in effect, work requests were issued, and TS requirements were being followed.

(1) WR/JO 95-ACAY1, Remove Reactor Head Before Core Offload

Prior to fuel offload, on September 11 the inspector observed personnel removing the integrated reactor vessel head and placing it on its storage stand located on the refueling floor. This evolution was performed in accordance with procedure CM-M0094, Integrated Reactor Vessel Head And Upper Internals Removal, by both licensee and contractor personnel. Personnel were stationed on the refueling operating deck and at the refueling cavity floor ensuring that the head lifted smoothly without snagging guide studs, the currently stuck stud, or upper internals components. Once the head was removed and the upper internals exposed, personnel quickly evacuated the refueling cavity in keeping with ALARA practices.

After plant personnel removed the head from the cavity and hoisted it over the operating deck en route to the storage stand, the inspector noted that no one verified that the IRVH was lifted a maximum of 12 inches above the floor as required by precautions in the procedure. It appeared to the inspector located across the refueling cavity approximately 50 feet away, that the head was slightly more than 12 inches off the ground. During the Spring 1994 refueling outage, contractor personnel lifted the head more than 2 feet off the ground to clear a handrail located on the refueling floor. Last year's action was contrary to the procedural precaution and resulted in an Adverse Condition Feedback Report. Because the head weighs approximately 180 tons, the 12-inch limitation was imposed for heavy load drop considerations. After the September 11 head-lift, the inspector discussed this year's observation with licensee management who documented it in a outage-improvement CR. Subsequent IRVH manipulations were performed with plant personnel verifying the head to be no more than 12 inches off the ground while outside the cavity.

The inspector also noted a few occurrences of inappropriate industrial safety acts such as individuals not tying off to safety ropes while standing near the cavity. These actions were also noted by the licensee's NAS inspector at the job site. The NAS inspector was aggressive in reminding workers of safety requirements. Despite the industrial safety incidents and the 12-inch verification issue, the inspector concluded that the overall head lift was done well.

(2) WR/JO 95-ADIS1, Replace Battery Bank 1A-SA

The inspectors observed activities while the licensee changed out the sixty-cell safety-related emergency battery 1A. The shop had previously sent personnel to the Robinson plant to gain experience by participating in a similar

activity there. Based on that experience, the shop had set up a small forklift in the battery room to move the cells between the rack and a conveyer extending from the doorway to an area clear of the switchgear. Battery cells were transferred between the conveyer and turbine building mezzanine using a small cart. The licensee had found, at the Robinson plant, that carts with a simple swivel front axle assembly tended to turn over, so they procured carts with tierod-type steering for the Harris changeout. Observed activities were being performed carefully and were well controlled. Subsequent testing is discussed in paragraph 4.b.(4) of this report.

The applicable vendor manual stated that, if lubrication was needed while sliding cells across the plastic tray rail covers, use plain unscented talcum powder. This was to preclude long term chemical reactions between powder ingredients and the polycarbonate battery cell cases. The inspector observed that, during removal of the old cells from the old rail covers, the licensee was using a barber shop type powder with a number of ingredients - including a fragrance. This had been provided by the system engineer. After notification, the shop ceased using it during the removal process and did not use powder while installing the new cells on top of new rail covers, thus reducing future operability uncertainty.

(3) WR/JO 95-ABUG5, Containment Equipment Hatch Closure Time Test

The inspectors observed the licensee reinstall the containment equipment hatch as a "time test" to establish a baseline for future outages when there may be a need to conduct mid-loop operations with the equipment hatch open. The equipment hatch had not been opened since original licensing because the personnel hatch was large and accommodated most equipment. During this outage, the equipment hatch was opened to pass RCP deck plugs and the reactor cavity permanent seal ring.

The hatch reinstallation was controlled by procedure CM-M0100, Rev. 3/2, Containment Equipment Hatch Removal and Replacement, sections 7.4 for immediate closure and 7.3 for normal closure. "Immediate closure" would result in the hatch being closed by 4 specific bolts (of 36). The post-trial assessment was very good and contained many good points for consideration. As expected for a first-time evolution, the test showed that minor coordination improvements could be readily made; that certain steps, such as platform removal, did not have back up provisions; and that using more than the TS-required 4 bolts might be necessary so that the remaining bolt holes would be

adequately lined up in case additional bolts or full closure were subsequently necessary. The licensee did not have a need for the "immediate closure" provisions during this outage. The inspectors had no further comments on this test.

In general, the performance of work was satisfactory with proper documentation of removed components and independent verification of the reinstallation. The inspectors identified no violations or deviations in this area.

b. Surveillance Observation (61726)

The inspector observed several surveillance tests to verify that approved procedures were being used, qualified personnel were conducting the tests, tests were adequate to verify equipment operability, calibrated equipment was used, and TS requirements were followed. During the recent refueling outage, the inspectors observed several 18-month TS surveillance tests, including the following:

(1) OST-1813, Rev. 5, Remote Shutdown System Operability.

This procedure partially satisfied requirements contained in TS 3/4.3.3.5, Remote Shutdown System. The procedure verified that transfer switches, Auxiliary Control Panel (ACP) controls, and Auxiliary Transfer Panel controls were operable for those components required by the SHNPP Safe Shutdown Analysis to remove decay heat, control RCS inventory through normal charging, control RCS pressure, control reactivity, and remove decay heat via the RHR system.

The inspector observed portions of Test C (section 7.3) of this procedure, which tested the "B" train transfer panels, TP-1BSB and ATP-1BSB, and associated ACP control switches. As directed by the procedure, operators placed each required transfer panel switch in the TRANSFER position and later verified that ACP control switches operated their respective safe shutdown components. Prior to the test performance, as an operator aid, plant personnel placed STAR placards around each control switch to be cycled, ensuring that operators cycled the right components per the procedure. While there were some unexpected occurrences during the procedure, including blown-out indicating light bulbs on the ACP, and unanticipated annunciators and actions regarding the ESCWS chiller condenser, the few obstacles were easily resolved and the test was successfully completed. The inspector noted good personnel performance during this surveillance test.

(2) OST-1823, Rev. 8, 1A-SA Emergency Diesel Generator Operability Test, 18 Month Interval, Section 7.4.

This test procedure partially satisfied 18 month surveillance requirements contained in TS 4.8.1.1.2f. Steps in section 7.4 of the procedure verified that, on a simulated loss of off-site power in conjunction with a safety injection test signal, the "A" emergency bus deenergized, load shedding occurred, the "A" EDG started, emergency busses were reenergized in a timely manner, and emergency loads were energized through the sequencer. An essential element of the test was operators simultaneously simulating containment spray actuation and SI signals, while manually tripping the normal electrical feed from auxiliary bus D to emergency bus A-SA.

During the initial performance of this test on September 3, operators erroneously performed the simulations first, then opened the normal circuit breaker. This caused the sequencer to start out on the incorrect sequence program, then reset and reinitiate using the correct program. The "A" RHR pump started twice in about a twenty second interval. The "A" containment spray pump failed to start the second time. The licensee determined that the test performance did not satisfy the procedure and that starting the RHR pump twice in twenty seconds was contrary to technical requirements in the manufacturer's manual. The licensee suspended the test until September 4.

Having suspended the test, the licensee evaluated the RHR pump condition in ESR 9500738, Revisions 0 and 1 and evaluated the test control situation in CR 95-2012. These evaluations concluded that the RHR pump had not been damaged but that the test should be rerun. The inspector reviewed the ESR and concluded that it reached correct conclusions.

On September 4 morning, the inspector observed the test being set up again. The prebriefing in the control room was well performed and personnel were dispatched to stations. The "B" RHR pump was already stopped and the "A" pump was stopped for the test, allowing the reactor core to heat up at 30 °F per hour. When the test participants did not report to their stations promptly, the SCO aborted the test, restarted A RHR pump to reestablish core cooling, and initiated management actions to improve personnel performance. The inspector noted that the SCO's response was prompt and appropriate. Test section 7.4 was successfully completed later that day with all components starting as expected.

(3) OST-1825, Rev. 8, Safety Injection: ESF Response Time, Train A 18 Month Interval On A Staggered Basis, Modes 5-6.

This test procedure satisfied various 18 month TS surveillance sections requiring safety-related equipment response to test signals. During the test, the inspector observed that components actuated as required. The near 200-page procedure contained some minor inadequacies, and there were some personnel errors noted during its performance. Some of these resulted in the procedure having to be temporarily revised so that missed elements of one procedure section could be captured in subsequent sections. In one case, containment fan cooler switches were placed in the wrong pretest position, preventing the fans from starting on the SI signal and requiring operators to reverify the fans would start in a later section. In another case, as discussed in paragraph 4.c below, a special temporary procedure (OST-9013T) had to be developed to retest certain AFW system valves. The inspector, however, considered the number of errors during OST-1825 performance to be minimal considering the procedure's complexity. Overall, licensee performance during this surveillance test was adequate. Additionally, plant equipment responded properly to the SI signals.

(4) MST-E0027, 1E Battery Service Test, [1A Battery]
MST-E0013, 1E Battery Performance Test, [1A Battery]
MST-E0012, 1E Battery 18-Month Test, [1A Battery]
CM-E0001, Station Battery Equalizing Charge, [1A Battery]

These procedures were associated with the installation and placing into service the replacement 1A-SA battery. The inspectors observed that procedures were well run and data was properly collected. During the first attempt to perform MST-E0027, shop personnel, recognizing that the computerized test equipment was not controlling the battery load properly, stopped the test, conferred with the test apparatus vendor, and corrected the setup. The test was subsequently performed without incident.

The inspectors found satisfactory surveillance procedure performance 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 observed no violations or deviations in this area.

c. Effectiveness of Licensee Control in Identifying, Resolving, and Preventing Problems (40500)

Condition Report CR 95-02534 was generated after an inadequate surveillance test procedure resulted in a partial safety injection on October 5. Temporary procedure OST-9013T, Temporary Procedure for Testing the MDAFW Pump FCVs Auto Open Feature From K635, was written to verify that the AFW flow control valves would open on a

safety injection signal. The valves' auto open feature was originally supposed to have been tested earlier in the week during the performance of safety injection test procedure OST-1825. However, because of an error in that procedure, operators did not verify the valves opened. Specifically, the procedure did not take into account the short duration (twenty seconds) of the valves' auto open signal. By the time OST-1825 directed operators to verify the valves opened, the twenty second signal was gone and the valves had reclosed.

Retest procedure OST-9013T on October 5 directed personnel to install jumpers and lift leads, ensuring all logic was satisfied to open the valves on a safety injection signal. Additionally, the lifted leads prevented the signal from actuating other safety injection components, such as the emergency safeguards sequencer. The procedure directed operators to install a switched jumper between two relay contacts which, upon closing the switch, would initiate a safety injection signal to the valve controllers. Operators writing the procedure did not understand that a latching relay in the circuit would necessitate temporarily installing another jumper during test restoration to remove the SI signal. The procedure writers assumed that simply opening the switch on the actuating jumper would reset SI. As directed by the procedure, after the valves were satisfactorily tested and with the safety injection signal still present, plant personnel relanded the previously lifted leads. This action started the "B" train Emergency Safeguards Sequencer and sent start signals to several components including the "B" RHR and CCW pumps. After consulting with cognizant engineers and associated drawings, personnel were able to remove the SI signal and secure the affected equipment. The inspectors noted that all equipment started as required following the partial SI. Additionally, no components were damaged following the unexpected actuation.

During subsequent discussions with licensee personnel, the inspector determined the root cause of this event to be inadequate procedures caused by the operators who wrote, reviewed, and approved the procedure failing to fully understand the equipment function.

10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. The licensee's Corporate Quality Assurance Program Manual, which implements the regulatory requirement, states in section 6.3.6 that provisions shall be made for the review of procedures required by plant commitment by an individual knowledgeable in the area affected to determine the need for changes. The failure to have fully knowledgeable procedure writers and reviewers for procedure OST-9013T is contrary to the above requirements and resulted in an unplanned and partial safety injection.

This is identified as Violation 400/95-15-02, Failure to Provide for the Review of a Safety Injection Test Procedure by an Individual Fully Knowledgeable in the System Logic.

The safety significance of this event was minor. The RHR pump operated in recirculation mode, ensuring minimum flow and preventing pump damage. The charging/safety injection pump was already operating for normal charging when the SI occurred. Operators verified that all equipment started as required on the SI signal, and were later able to reset SI. At the close of the inspection period, the licensee was still developing corrective actions.

This event was properly reported to the NRC in accordance with 10 CFR 50.72 requirements. A 30 day LER was pending at the end of the inspection period.

Overall, one violation was identified in the maintenance and surveillance area. Otherwise, surveillance and maintenance procedures were performed effectively with only minor issues requiring resolution.

5. ENGINEERING

a. Design and Installation of Plant Modifications (37551)

ESRs involving the installation of new or modified systems were reviewed to verify that the changes were reviewed and approved in accordance with 10 CFR 50.59, that the changes were performed in accordance with technically adequate and approved procedures, that subsequent testing and test results met approved acceptance criteria or deviations were resolved in an acceptable manner, and that appropriate drawings and facility procedures were revised as necessary. The licensee was challenged with many issues requiring engineering resolution during the outage. The following engineering evaluations, modifications, and/or testing in progress were inspected.

(1) ESR 9400013, Permanent Cavity Seal Ring

This design package provided engineering instructions for installing the new permanent refueling cavity seal ring. For previous refueling outages, the licensee installed a temporary seal ring with inflatable bladders which consumed a lot of outage time and personnel dose. The new seal ring, which would be permanently welded to the cavity and reactor vessel, would eliminate recurring installation and removal hassles. The new design would include several removable hatch covers allowing proper ventilation for RCS components located in the reactor cavity annulus area.

Due to fitup problems, the new seal ring project was abandoned and the licensee resorted to using the old inflatable seal ring for cavity floodup prior to core reload during this outage. Specific problems centered around the seal ring not fitting properly between the reactor vessel seal ledge and the cavity liner. Additionally, there were problems achieving proper clearances between the three major sections of the ring. Proper clearances were necessary for welding the three sections together and welding the ring to the cavity liner and vessel ledge. Following the unexpected installation delays and eventual project abandonment, the licensee initiated a Condition Report and a root cause investigation into contributing factors.

The inspectors observed part of the attempted installation, reviewed portions of the design package, and concluded that neither poor workmanship nor poor engineering contributed to the project failure. The design package was adequately detailed with specific installation instructions, adequate drawings, and a good unreviewed safety question determination. With a piece of equipment as large and complex as the new cavity seal ring, which was manufactured offsite by a vendor, fitup problems like these were possible. Licensee management informed the inspector that the permanent seal ring installation would be attempted again during the next refueling outage.

(2) ESR 9500876, Stuck Reactor Vessel Closure Stud #40

This ESR documented the acceptability of a reactor vessel head stud #40 which became stuck during reinstallation on October 2. When the stud became stuck, personnel could not obtain full stud engagement into the reactor vessel. Actual stud engagement was approximately 6.625 inches vice the standard of 7.375 inches for 5-13/16 inch diameter studs. The engineering evaluation considered the engagement deviation and the fact that the stuck stud could not be removed for required ASME Section XI inspection. From calculations, engineers determined that the stud could still be normally tensioned along with the remaining studs, and the stresses, either from the tensioning or the pending RCS heatup, would not exceed effective code allowables. Additionally, since the 10-year inspection would not be due until 1997, the stud could remain in the vessel flange during the upcoming operating cycle. The inspector considered the evaluation to be adequate.

(3) ESR 9500738, Rev. 0 and Rev. 1, RHR Pump A Starting Frequency

This ESR evaluated the effect of having the "A" RHR pump start twice in twenty seconds during the OST-1823

performance discussed in paragraph 4.b.(2) of this report. The main concern was whether or not the pump starts imposed excessive heat stresses on the motor. The evaluation roughly equated the rate of heat production during the first stages of motor acceleration to that generated in a stall condition. Engineers then obtained the maximum safe stall time (ten seconds) for the RHR pump motor from the associated plant specification. Plant engineers assumed that heat stresses on the motor in a stall event would envelope the heating experienced in an acceleration event of equal duration. With the acceleration time for the pump motor being 0.8 seconds, two start events would equal 1.6 seconds which was within the ten second maximum safe stall time. Thus, the evaluation concluded that the motor was not overstressed by the two starts. Revision 0 of the ESR contained an error regarding the time between the two starts. Plant personnel detected the error and corrected it in Rev. 1. The inspector considered this evaluation to be adequate.

(4) ESR 9500752, CSIP 1B-SB Safety Significance of Stuck Miniflow Check Valve

As discussed in paragraph 3.a.(1) of this report, the "B" CSIP miniflow check valve 1CS-193 failed a forward flow test due to a stuck disk inside the valve. One of the valve's safety functions was to pass at least 60 gpm forward flow during a safety injection to prevent pump damage. The "B" pump was erroneously placed in service for seven hours during Mode 4 on September 2 with the valve in the degraded condition. The ESR analyzed the condition for a postulated LOCA and inadvertent SI. These were the most limiting events possibly requiring the miniflow system to function with the plant in Mode 4 and RCS pressure reduced to approximately 350 psig. After considering the plant operating conditions at the time and the availability of other systems, the evaluation concluded that the pump would have performed its intended safety injection function and would not have been damaged without the miniflow system available. The evaluation further showed that the pump never operated in Mode 4 on September 2 pumping less than the 60 gpm minimum recommended by the pump's vendor. The inspector considered this evaluation to be adequate. As mentioned in report paragraph 3.a.(1), the licensee reported the check valve deficiency in accordance with 10 CFR 21.21.

(5) ESR 9500098, Rev. 0, Closed Cell Tubing/Sheet Type Insulation Evaluation.

This ESR evaluated the use of closed cell tubing and sheets for anti-sweat insulation for essential and nonessential chilled water piping greater than five inches in diameter.

This had been previously approved by licensee engineers in PCR 7346 (October 1994) for pipe less than five inches diameter. This subject had attracted the inspectors attention when they happened upon a "Transient Combustible" tag hanging on a permanent insulation installation in August 1995.

The ESR specifically evaluated two aspects of the insulation. First was thermal conductivity, which was about twice as high as the currently used fiberglass insulation. Heat gain from the room would be offset by a lower load on the HVAC units cooling the room. The inspector concluded that this was a reasonable approach that demonstrated understanding of the systems and building environment. Second was the change in fire loading caused by changing from fiberglass material to hydrocarbon (plastic) material. The ESR concluded that the material was not combustible based on statements in NRC BTP CMEB 9.5-1. This meant that room fire loading would not require updating when the material was installed. In contrast to this evaluation, the licensee's applicable Chemical and Consumables Fact Sheet, AP-501-01059 Revision 1, stated that the material was combustible. That was the reason for the "Transient Combustible" tag hanging on a permanent insulation installation. The inspector reviewed the relevant literature and concluded that the ESR conclusion was flawed. The BTP defines a number of noncombustible materials and then states that they may have a thin coating, not over 1/8 inch thick, of material with fire properties similar to this. This material is about one inch thick, not fitting the BTP exception. Thus, this material was clearly flammable. When informed of the NRC conclusion, the licensee proceeded to recalculate the fire loading of affected locations.

(6) ESR 9500809, As-found Set Pressure of Valve 1RC-127 out of Tolerance

This evaluation evaluated the acceptability of valve test results from September 13 which were outside of set pressure acceptance criteria. Specifically, pressurizer code safety valve 1RC-127 is required by TS 3.4.2.2 to be operable having a lift setpoint of 2485 psig with a margin of plus or minus one percent. The upper margin limit equates to 2509.85 psig. The valve was shipped to a contract laboratory and later failed its initial test by lifting at 2516 psig. Three subsequent tests revealed lift pressures of 2493, 2487, and 2482 psig.

The licensee's evaluation stated that the initial failure was probably caused by shipping problems. This was supported by the three consecutive acceptable tests.

Nevertheless, the evaluation appropriately considered the effects of the valve lifting at 2516 psig during the most limiting overpressurization design basis accident (a main turbine trip). The evaluation showed that if the valve lifted at 2516 psig, RCS pressure would still be maintained less than 110 percent of design (2750 psig) with margin remaining. The evaluation took credit for proper operation of the other two safety valves. The inspectors considered this evaluation to be adequate.

The inspectors identified no violations or deviations in the design/installation/testing of modifications area.

b. Review of LERs (92700)

(Open) LER 95-006-00, Emergency Core Cooling System Piping Not Fully Contained Within Reactor Auxiliary Building Emergency Exhaust System Boundary, Resulting in Condition Outside Design Basis.

This LER discussed the Reactor Auxiliary Building Emergency Exhaust System construction deficiency which resulted in certain portions of the ECCS being outside of the emergency ventilation boundary. This issue is discussed in more detail in NRC IR 400/95-13. The LER will remain open pending licensee completion and inspector review of the root cause investigation and associated corrective actions. The licensee intended to supplement the LER when the root cause investigation was completed.

Overall, engineering activities were performed well, especially considering the many engineering challenges that were presented during the refueling outage. The inspectors identified no violations or deviations in the engineering area.

6. PLANT SUPPORT

a. Plant Housekeeping Conditions (71707) - The inspectors reviewed storage of material and components, and observed cleanliness conditions of various areas throughout the facility to determine whether safety hazards existed. With the plant in a refueling outage, much of the plant was staged at various times with scaffolding for maintenance activities. At various times, unused scaffolding was stored throughout the plant in designated storage locations. Plant equipment that was not part of some work scope was in generally good condition. Overall, plant equipment and overall housekeeping was adequate during the outage. The inspectors observed no safety hazards.

b. Radiological Protection Program (71750) - The inspectors reviewed radiation protection control activities to verify that these activities were in conformance with facility policies and

procedures, and in compliance with regulatory requirements. The inspectors also verified that selected doors which controlled access to very high radiation areas were appropriately locked. Radiological postings were likewise spot checked for adequacy.

Since this was an outage month, the inspectors particularly focused on licensee controls for containment entries and hot work. A containment entry window was manned around the clock by HP personnel keeping track of workers inside containment. For major jobs like the IRVH lift and other core manipulations, HP personnel were present, ensuring worker exposure was minimized. The licensee spent less worker dose during the initial vessel head lift than in any year before. Operators made good use of remote dosimetry monitoring technology for the steam generator jobs. To better control worker exposure, the licensee established communications between workers in high exposure areas and HP personnel in remote locations. The licensee bettered its dose goal by over 16 person-rem, reaching 142.7 person-rem vs. a goal of 159.3 person-rem. Unfortunately, the number of personnel contamination events (PCEs) during the outage exceeded the licensee's outage goal by 22 (122 vs. 100). Although the licensee investigated each PCE as it occurred, the licensee's investigation into the excessive total was continuing at the end of the inspection. Each PCE was appropriately documented on a condition report.

Overall, the inspector considered outage performance in radiological controls to be good.

c. Security Control (71750) - During this period, the inspectors toured the protected area and noted that the perimeter fence was intact and not compromised by erosion or disrepair. The fence fabric was secured and barbed wire was angled. Isolation zones were maintained on both sides of the barrier and were free of objects which could shield or conceal an individual. The inspectors observed various security force shifts perform daily activities, including searching personnel and packages entering the protected area by special purpose detectors or by a physical patdown for firearms, explosives, and contraband. Other activities included vehicles being searched, escorted, and secured; escorting of visitors; patrols; and compensatory posts. In conclusion, the inspectors found that selected functions and equipment of the security program complied with requirements.

d. Fire Protection (71750) - The inspectors observed fire protection activities, staffing and equipment to verify that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable. During plant tours, the inspector looked for fire hazards. The inspector concluded that the fire equipment and barriers inspected were in proper physical condition. As stated in paragraph 3.e of

this report, plant response to a perceived switchgear fire was excellent.

e. Emergency Preparedness (71750) - The inspectors toured emergency response facilities to verify availability for emergency operation. Duty rosters were reviewed to verify appropriate staffing levels were maintained. As applicable, the inspectors observed emergency preparedness exercises and drills to verify response personnel were adequately trained. No emergency preparedness exercises or drills were performed during this inspection period.

f. Plant Nuclear Safety Committee Meetings (40500) - The inspectors attended several PNSC meetings during the refueling outage to verify effectiveness of the licensee's onsite safety committee. These meetings discussed a wide range of plant issues including some discussed earlier in this report: midloop operations, containment hatch closure, and a reactor vessel stuck stud. The inspector noted that a PNSC quorum was always present and that qualified individuals were on the committee. The inspector reviewed selected meeting minutes and determined that they were adequate. The inspector noted that committee members had the necessary safety focus during the meetings.

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, emergency preparedness requirements, and TS requirements in these areas was satisfactory. The inspectors identified no violations or deviations in the plant support area.

7. EXIT INTERVIEW

The inspector met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on October 6, 1995. 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 and LERs 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	Status	Description and Reference	
95-015-01	Open	VIO	Failure to Properly Annotate Surveillance Test Requirement for an Inoperable CSIP, paragraph 3.a.(1).
95-015-02	Open	VIO	Failure to Provide for the Review of a Safety Injection

			Test Procedure by an Individual Fully Knowledgeable in the System Logic, paragraph 4.c.
95-011-01	Closed	IFI	Questionable Position Indication for Containment Isolation Valve 1SP-209, paragraph 3.g.
95-006-00	Open	LER	Emergency Core Cooling System Piping Not Fully Contained Within Reactor Auxiliary Building Emergency Exhaust System Boundary, Resulting in Condition Outside Design Basis, paragraph 5.b.
95-007-00	Open	LER	Inadvertent Start of the Turbine Driven AFW Pump/Unplanned ESF Actuation and Identification of an Additional Related Test Deficiency, paragraph 3.f.(1).
95-008-00	Closed	LER	"B" Charging/Safety Injection Pump was Returned to Service Prior to Required Acceptance Testing, Resulting in Technical Specification Violation, paragraph 3.f.(2).

8. ACRONYMS AND INITIALISMS

ACP - Auxiliary Control Panel
AFW - Auxiliary Feedwater
ALARA - As Low as Reasonably Achievable
ASME - American Society of Mechanical Engineers
BTP - Branch Technical Position
CCW - Component Cooling Water
CFR - Code of Federal Regulations
CM - Corrective Maintenance [procedure]
CP&L - Carolina Power & Light
CR - Condition Report
CSIP - Charging/Safety Injection Pump
CVCS - Chemical and Volume Control System
ECCS - Emergency Core Cooling System(s)
EDG - Emergency Diesel Generator
EIR - Equipment Inoperable Record
encl - Enclosure
ESCWS - Essential Services Chilled Water System
ESF - Engineered Safeguards Feature

ESR	-	Engineering Service Request
ESW	-	Emergency Service Water
°F	-	Degrees Fahrenheit
FCV	-	Flow Control Valve
FHP	-	Fuel Handling Procedure
FME	-	Foreign Material Exclusion
FR	-	Federal Register
GP	-	General Procedure
GPM	-	Gallons Per Minute
HVAC	-	Heating, Ventilation, and Air Conditioning
IFI	-	Inspector Followup Item
IR	-	[NRC] Inspection Report
IRVH	-	Integrated Reactor Vessel Head
ISI	-	Inservice Inspection
LCO	-	Limiting Condition for Operation
LER	-	Licensee Event Report
LOCA	-	Loss of Coolant Accident
LTOP	-	Low-Temperature Overpressure Protection
MDAFW	-	Motor-Driven Auxiliary Feedwater [pump]
MST	-	Maintenance Surveillance Test [procedure]
NAS	-	Nuclear Assessment Section
NPF	-	Nuclear Production Facility [a type of license]
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NUREG	-	NRC Technical Report Designation
OMM	-	Operations Management Manual
OST	-	Operations Surveillance Test [procedure]
PCE	-	Personnel Contamination Event
PCR	-	Plant Change Record
PDR	-	Public Document Room
PIG	-	Particulate, Iodine, and Gas [monitor]
PNSC	-	Plant Nuclear Safety Committee
PORV	-	Power-Operated Relief Valve
psig	-	Pounds per Square Inch, Gauge
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RHR	-	Residual Heat Removal
RII	-	Region Two [NRC Office]
SCO	-	Senior Control Operator
SG	-	Steam Generator
SI	-	Safety Injection
TADOT	-	Trip Actuating Device Operational Test
TDAFW	-	Turbine-Driven Auxiliary Feedwater [pump]
TS	-	Technical Specification [part of the facility license]
VIO	-	Violation [of NRC requirements]
vs	-	Versus
WR/JO	-	Work Request/Job Order