U.S. NUCLEAR REGULATORY COMMISSION (NRC)

REGION II

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Report No	50-348/97-11 and 50-364/97-11
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Farley Nuclear Plant (FNP), Units 1 and 2
Location:	7388 North State Highway 95 Columbia, AL 36319
Dates:	September 7 through October 18, 1997
Inspectors:	T. Ross, Senior Resident Inspector (SRI) J. Bartley, Resident Inspector (RI) R. Caldwell, RI J. Canady - RI from Hatch Nuclear Plant (Sections M8.1 and M8.2) N. Merriweather, Region II Reactor Inspector (Sections E8.4 - E8.25) J. Zimmerman, NRR (Section E1.2)
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Enclosure 2

EXECUTIVE SUMMARY

Farley Nuclear Power Plant, Units 1 and 2
NRC Inspection Report 50-348/97-11, 50-364/97-11

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of onsite resident inspector and region-based specialist inspections.

Operations

- Operator attentiveness to main control board (MCB) annunciator alarms and response to changing plant conditions were prompt. Management's persistent efforts to reduce the number of MCB deficiencies and achieve "blackboard" were evident. Operating crews demonstrated a high level of awareness of plant conditions and ongoing activities (Section O1.1).

- Control Room professionalism, operator demeanor, teamwork, and conduct of business in the main control room were appropriate and effective. Shift supervisor (SS) command and control functions and operations management oversight were evident (Section O1.1).

- Overall material conditions of Unit 1 and 2 structures, systems and components (SSCs) were good. Almost all plant areas were clear of trash and debris. Areas inside Unit 2 containment were in satisfactory condition (Section O2.1).

- Safety system walkdowns and tours verified that accessible portions of selected systems were adequately maintained and operational (Sections O2.1 and O2.2).

- Operations had become complacent in its implementation of the Night Orders Book (NOB). It was evident that some SSs were not always reviewing the NOB in a timely manner (Section O3.1).

- Operations management implemented immediate and effective compensatory measures for Unit 1 by establishing more restrictive administrative controls over primary coolant specific activity to address increased projected end-of-cycle (EOC) steam generator (SG) conditional tube leakage (Section O4.1).

Maintenance

- Maintenance and surveillance testing activities were generally conducted in a thorough and competent manner by qualified individuals in accordance with plant procedures and work instructions (Section M1.1).

- Maintenance and support activities associated with the replacement of PT456, Pressurizer Pressure Channel 2, were generally well-controlled, and performed by competent and experienced personnel. A technical issue concerning the adequacy of response time testing and the capability of Foxboro transmitters to respond to high ramp rates was identified (Section M1.2).

- While the calibration and RTT associated with the replacement of PT456 were being performed, the QC supervisor recognized that the transmitter was not EQ based on the purchase order number. (Section M1.2)

- Design change for R11/12 Containment Air Particulate and Gas Radiation Monitors was properly implemented and exhibited good craftsmanship (Section M1.3).

- Corrective actions were properly identified and satisfactorily implemented for several Licensee Event Reports, and NRC violations and open items (Section M8.1 through M8.7).

Engineering

- Engineering test procedure for fully withdrawing Unit 2 control rods to a new position was well written and controlled. The evolution was conducted in a smooth and deliberate manner (Section E1.1).

- The NRC staff found Southern Nuclear Operating Company's (SNC's) "Generic Letter 95-05 Safety Assessment" in response to Generic Letter 95-05 requirements to be adequate. Compensatory actions were appropriate, and reporting requirements were adequately addressed (Section E1.2).

- Overall, licensee response to identified service water pipe corroded conditions was prompt and effective (Section E1.3).

- Resolution of Updated Final Safety Analysis Report (UFSAR) discrepancy #089 were not thorough. Calculations supporting the design of the air start system were non-conservative and were not validated against existing air start test data. (Section E8.1).

- A violation was identified for not performing reverse flow testing of the turbine driven auxiliary feedwater (TDAFW) discharge check valve V003 or V002D, F, and H to verify that the disk travels to the seat on cessation or reversal of flow (Section E8.6).

- A violation was identified for failure to recognize a TS change was required for the safety evaluations performed for changes to the Auxiliary Building Battery Service Test Procedure FNP-1(2)-STP.905.1 and UFSAR Section 8.3.2 associated with PCN B-92-0-8099 to include the results of Calculation 07597-E144 addressing design basis requirements for battery duty cycle, load profile and voltage requirements (Section E8.22).

- A violation was identified for failure to have a test program and procedures for service testing of the TDAFW Class 1E battery to ensure that the battery would meet the required battery duty cycle (Section E8.8).

- A violation was identified for inappropriate acceptance criteria in the surveillance procedure for verifying the forward flow of check valve V003 and for failure to follow drawings and instructions in the installation of the Unit 2 TDAFW battery structural/electrical component installations (Section E8.7 and E8.9).

- A violation was identified because design control measures did not ensure that calculations were verified and controlled adequately (Section E8.16 and E8.25).

- A violation was identified for failure to correct a deficiency identified during the 1990 Safety System Self-Assessment (SSSA) of the Component Cooling Water (CCW) system involving differences between Piping and Instrumentation Drawings (P&IDs) and procedures in identifying caps on vent and drain lines (Section E8.20).

Plant Support

- On occasion, operators demonstrated poor work practices in applying protective actions to prevent the inadvertent spread of contamination during partial entries into contaminated areas. (Section R1.1).

- Overall cleanliness and housekeeping of radiologically controlled areas (RCA) of the auxiliary building was good. Ongoing decontamination efforts by the Health Physics (HP) department to reduce contaminated surface areas were aggressive and continue to be successful (Section R2.1).

- HP actions to address long-term storage of spent resins were prompt and thorough (R8.1).

- An announced drill of the licensee's emergency plan was considered to be reasonably challenging. Response facilities were manned and fully operational in a timely manner. Emergency response personnel properly characterized evolving events and made accurate and timely emergency classifications and notifications (Section P1.1).

- Security personnel activities observed during the inspection period were performed well. Site security systems remained functionally adequate to ensure physical protection of the plant (Section S1.1).

- A violation was identified for failure to adequately control and mark the Safeguard Information (SGI) maintained in the at-the-controls area of the Main Control Room (Section S3.1).

Report Details

Summary of Plant Status

Unit 1 operated continuously at 100% power for the entire inspection period.

Unit 2 operated continuously at 100% power for the entire inspection period. On October 15, the unit reached 300 days of continuous operation at power.

I. Operations

O1 Conduct of Operations

O1.1 Routine Observations of Control Room Operations

a. Inspection Scope (Inspection Procedure (IP) 71707)

Inspectors conducted frequent inspections of ongoing plant operations in the Main Control Room (MCR) to verify proper staffing, operator attentiveness, adherence to approved operating procedures, communications, and command and control of operator activities. Inspectors reviewed operator logs and Technical Specification (TS) Limiting Conditions of Operation (LCO) tracking sheets, walked down the Main Control Boards (MCBs), and interviewed members of the operating shift crews to verify operational safety and compliance with the TSs. The inspectors frequently attended morning plant status meetings and shift turnover meetings to maintain awareness of overall facility operations, maintenance activities, and recent incidents. Morning reports and Occurrence Reports (ORs) were reviewed on a routine basis to assure that the licensee properly reported and resolved potential operational safety concerns.

b. Observations and Findings

Overall control and awareness of plant conditions during the inspection period remained a strength. Inspectors observed that the Unit 1 MCB, Unit 1 Balance of Plant (BOP), and the Emergency Power Board (EPB) annunciator alarm panels were frequently "blackboard." However, the Unit 2 MCBs and BOP panels continued to have a few persistent annunciators for known equipment problems. Management efforts to maintain MCB deficiencies at very low levels and blackboard conditions continued. The combined MCB deficiencies on Units 1 and 2 dropped to ten, the lowest level for several months. Only two of the MCB deficiencies were on Unit 1. The majority of the deficiencies involved non safety-related instrumentation or equipment, and none resulted in a TS LCO. Tagging and work control activities were conducted outside the at-the controls (ATC) area of the MCR. Access to the ATC area was controlled to limit unnecessary activities.

Operator attentiveness to MCB annunciator alarms and response to changing plant conditions remained effective. Interviews with members of the operating crew verified that they were consistently aware of plant conditions and ongoing activities. There were no challenging incidents or transients necessitating operator response during the report period. Steady-state operations of both units were well-controlled and continued without any significant events. Operator logs were of sufficient detail and scope. Shift staffing was verified to be in compliance with procedural and TS requirements. Pre-shift briefings of the operating crews by the shift supervisors (SS) were generally concise and provided operators with shift direction and priorities. Shift turnovers were accomplished in an orderly manner, following a board walkdown by the off-going and on-coming operators and SSs.

c <u>Conclusions</u>

Control Room professionalism remained good. Operating crew demeanor, team work, and conduct of business were effective. Unit SS command and control, and operations management oversight were evident.

Operator attentiveness to MCB annunciator alarms and response to changing plant conditions were prompt. Management's persistent efforts to reduce the number of MCB deficiencies were evident. The operating crew consistently demonstrated a high level of awareness of existing plant conditions and ongoing plant activities.

O2 Operational Status of Facilities and Equipment

O2.1 <u>General Tours of Specific Safety-Related Areas (IP 71707)</u>

General tours of safety-related areas were performed by the inspectors throughout both units to examine the physical condition of plant equipment and structures, and to verify that safety systems were properly aligned. These general walkdowns included the accessible portions of safety-related structures, systems, and components (SSC).

Overall material conditions of Unit 1 and 2 SSCs were good. Almost all plant areas were clear of trash and debris. Some minor equipment and housekeeping problems identified by the inspectors during their routine tours were reported to the responsible SS and/or maintenance department for resolution. None of the problems constituted an immediate safety or compliance issue. However, some of the more significant findings identified by the inspectors during routine plant tours did require prompt response by the licensee, as follows:

1) The inspectors toured the Service Water Intake Structure (SWIS) with the Team Leader (TL) responsible for the painters. The inspectors pointed out the examples of the poor painting preparation referenced in Inspection Report (IR) 50-348, 364/97-10. The TL concurred with the inspectors assessment. The licensee's staff is evaluating methods to remove the old paint and corrosion products from these areas for proper preservation and painting. During the tour, inspectors also identified significant external corrosion (i.e., rust) on the 42-inch diameter Service Water System (SWS) discharge piping where it penetrated the north wall of the SWIS. The piping was subsequently examined by Engineering Support (ES) personnel (see report section E1.3 for details), and properly cleaned and painted.

2) On September 19, during a routine tour of the Emergency Diesel Generators (EDGs), the inspectors identified that the locking tabs for the 1B EDG fuel rack jam nuts for cylinders 1, 2, 3, 4, 5, 7, 8, 9, and 10 were not engaged. The inspectors verified that the lock tabs were engaged on the 1-2A and 2B EDGs. The inspectors immediately informed an SS. Occurrence Report (OR) 1-97-356 was generated to document the issue. The licensee promptly evaluated the deficiency, checked the jam nuts to ensure that they were not loose, and engaged the locking tabs. The licensee concluded that EDG operability was not impacted due to finding the jam nuts tight prior to engaging the locking tabs.

3) During a routine tour on September 12, the inspectors identified that an amphenol connection on the Unit 2 radiation monitor R29B was disconnected. This lead provided power to the check source mechanism for iodine detector Channels 3 and 4. The inspector identified this to the SS and the lead was subsequently reinstalled. The inspector reviewed Updated Final Safety Analysis Report (UFSAR) Section 11.4, "Process and Effluent Radiological Monitoring Systems," and found that paragraph 11.4.4.3 stated that these radiation monitors would be source checked on a monthly or quarterly basis. Although all of the TS required radiation monitors were being regularly source checked, neither the inspectors nor the licensee could identify any procedures requiring a monthly or quarterly source check of R-29A/B. This UFSAR discrepancy was not previously identified by the licensee's UFSAR reverification program. By the end of the report period, the licensee was still investigating the need to conduct regular source checks.

4) Although within TS required limits for level, the inspectors questioned whether the 1A Accumulator water level was decreasing at a faster rate than the other accumulators. Operations personnel performed a level trend analysis and were unable to account for approximately 30 to 40 gallons of accumulator water. At the next opportunity, Operations planned a containment entry to investigate a possible slow leak.

5) Although previously identified as a deficient condition, the inspectors found that leakage from the 1C Component Cooling Water (CCW) pump casing vent had increased significantly from its original 1 drop per minute (dpm) to 4-5 dpm. This increased leakage resulted in considerably more uncontrolled wetting of the pump skid surfaces with toxic, potentially contaminated chromated water. After notifying the SS, the inspectors observed that a catch device was promptly installed.

While at power, a limited tour of the Unit 2 containment was conducted on October 3, 1997, in conjunction with a job to replace one of the pressurizer pressure transmitters. The containment areas toured were in satisfactory condition.

02.2 Biweekly Inspections of Safety Systems (IP 71707)

Inspectors verified the operability of the following selected safety systems and/or equipment:

- Unit 2 High Head Safety Injection (HHSI) System, A and B Train
- Unit 1 and 2 Residual Heat Removal (RHR) System, A and B Train

Accessible portions of the systems listed above were verified to be properly aligned. The inspectors also observed them to be adequately maintained and in good operating condition. The inspectors did not identify any issues that adversely affected system operability. Minor deficiencies noted were discussed with the appropriate SS. The licensee's work to reduce the amount of potentially contaminated areas has significantly improved radiological conditions in the Unit 1 A and B Train RHR Pump Rooms. The majority of each room has been reclaimed which allowed routine touring without donning protective clothing. Decontamination efforts on the Unit 2 RHR pump rooms were in-progress. These decontamination efforts were considered a proactive and positive radiological practice.

02.3 Verification of Safety Tagging

a. Inspection Scope (IP 71707)

The inspectors verified that selected tagouts were implemented in accordance with procedural requirements. The inspectors reviewed and walked down selected devices tagged by the following tag orders (TOs):

- TO# 97-2283 Unit 1 Radiation Monitors R11 and R12
- TO# 97-2387-1 1B Emergency Air Compressor

b. Observations and Findings

The inspectors verified that devices identified on the tag orders were properly tagged. The device identifications were correct, tags were conspicuously placed on the devices and the tags did not obscure control room panel indications. The administrative aspects of filling out the tagging order forms were complete and correct. The tags placed were adequate for personnel safety and equipment protection.

c. Conclusion

The inspectors concluded that the reviewed safety tagging activities were correct and met the procedural requirements.

02.4 TS LCO Tracking (IP 40500 and IP 71707)

The inspectors routinely reviewed the TS LCO tracking sheets filled out by the shift foremen. All reviewed tracking sheets for Units 1 and 2 were consistent with plant conditions and TS requirements.

03 Operations Procedures and Documentation

03.1 Night Orders (IP 71707)

Administrative procedure FNP-0-AP-16, Conduct of Operation - Operations Group, Revision 27, Section 6.2, establishes general requirements for the Night Orders Book (NOB) maintained in the ATC area of the MCR. Occasionally, Operations management issued night orders for the SSs to read and implement as appropriate. Resident inspectors routinely review the NOB for new entries. In the past, inspectors observed that the SSs were very conscientious about initialing new night orders, acknowledging that they had read and understood the entry. However, the inspectors recently observed that most of the SSs were no longer initialing new entries in the NOB. Although AP-16 did not require initialing of night orders, this has been considered a good practice in the past. Furthermore, based on inspector interviews with SSs during the week of September 15, it became evident that some SSs were not always reviewing the NOB in a timely manner (even when a specific entry that expressly requested that it be reviewed prior to going on-shift). The inspectors discussed the use and purpose of the NOB with Operations management to better understand its role and management expectations. After these discussions, the NOB was reorganized to improve its useability and SSs were coached regarding its purpose.

04 Operator Knowledge and Performance

04.1 Administrative Limit For Primary Coolant Activity (IP 71707)

On September 11, Southern Nuclear Operating Company (SNC) held a conference call with the NRC to discuss its latest results regarding end-of-cycle (EOC) steam generator (SG) conditional tube leakage and burst calculations using data from the last Unit 1 refueling outage (U1RF14). During this call, SNC concluded that it was required by Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," Section 6, to notify the NRC that the EOC accident leakage exceeded the site-allowable leakage limit for Unit 1. By letter dated September 12, SNC submitted its GL 95-05 safety assessment and compensatory measures (see report paragraph E1.2 below).

Also during the conference call, SNC committed to implement immediate compensatory measures for Unit 1 by establishing more restrictive administrative controls over primary coolant specific activity. These administrative controls would limit the specific activity of primary coolant to 0.15 microcurie per gram dose equivalent I-131 (DEI) for steady-state conditions, and to 9 microcurie per gram DEI for transient conditions. The new limits are one-half of the limits specified by TS 3.4.9, Specific Activity. On the following day, the inspectors verified that the NOB contained an entry regarding the new administrative limit for primary coolant specific activity. The inspectors also interviewed the Unit 1 day-shift SS regarding his knowledge of the new DEI limits. The inspectors also verified that all oncoming Unit 1 and 2 licensed operator crews were subsequently briefed on the new administrative limits prior to assuming on-shift duties. SNC's commitment to implement the more restrictive administrative limits did not apply to Unit 2 until November 1, 1997.

By letter dated September 17, SNC submitted a license amendment for Units 1 and 2 to revise TS 3.4.9 to make it consistent with the more restrictive administrative limits for primary coolant specific activity.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (IP 61726 and IP 62707)

Inspectors observed and reviewed portions of various licensee corrective and preventive maintenance activities, and witnessed routine surveillance testing to determine conformance with plant procedures, work instructions, industry codes and standards, TSs, and regulatory requirements. The inspectors observed all or portions of the following maintenance and surveillance activities, as identified by their associated work order (WO), work authorization (WA), or surveillance test procedure (STP):

- FNP-2-STP-80.1 — 2B EDG Operability Test, Revision 24
- FNP-1-SOP-7.0A — Residual Heat Removal System
- FNP-1-SOP-7.0 — Residual Heat Removal System
- WO#S00079791 — Perform DCP S96-2-9060 on U2 R11/12
- FNP-1-STP-201.18 — Reactor Coolant System TE-412A and TE-412D Loop Calibration and Functional Test, Revision 39
- FNP-2-STP-256.4 — Pressurizer Pressure Sensor Response Time Test, Revision 5
- WO#M97006793 — 1B Emergency Air Compressor
- WO#00487300 — Calibrate Train B Differential Pressure Transmitter per FNP-1-IMP-218.2, Control Room Differential Pressure PDT-2768B, Revision 8
- FNP-1-STP-226.1 — BIG Sequencer Operability Test, Revision 6
- WA#W00482489 — 1B Auxiliary Building Battery Equalization per FNP-1-EMP-1341.08, Revision 3
- FNP-1-STP-11.2 — 1B RHR Pump Quarterly Inservice Test, Revision 32
- WA#W00483549 — Preventative maintenance task on 2A Boric Acid Tank temperature indication and alarm

b. Observations, Findings and Conclusions

All observed maintenance work and surveillance testing was performed in accordance with work instructions, procedures, and applicable clearance controls. In general, safety-related maintenance and surveillance testing evolutions were well-planned and executed. Responsible personnel demonstrated familiarity with administrative and radiological controls. Surveillance tests of safety-related equipment were consistently performed in a deliberate step-by-step manner by personnel in close communication with the Main Control Room (MCR). Overall, operators, technicians, and craftsman were observed to be knowledgeable, experienced, and well trained for the tasks performed.

M1.2 Replacement of Unit 2 Pressurizer Pressure Transmitter

a. Inspection Scope (IP 62707 and IP 61726)

The inspectors observed maintenance and surveillance activities associated with the replacement of Q2B31PT456, Pressurizer Pressure Channel 2. Specific activities included observation of time response testing and calibration of the new transmitter, observation of various pre-job briefings, reviews of completed test and calibration data sheets, and an at-power containment entry to observe installation of the new transmitter. The inspectors reviewed FNP-2-STP-256.4, "Pressurizer Pressure Sensor Response Time Test," Revision 5, FNP-0-IMP-430.16, "Environmentally Qualified Instrument Replacement Procedure," Revision 11, FNP-2-STP-201.5, "Pressurizer Pressure PT-456," Rev. 22, Updated Final Safety Analysis Report (UFSAR) Section 15.4, "Condition IV - Limiting Faults," UFSAR Section 7.2, "Reactor Trip System," Westinghouse WCAP-13632, "Elimination of Pressure Sensor Response Time Testing Requirements," Rev. 2, Electric Power Research Institute (EPRI) Report NP-7243, "Investigation of Response Time Testing Requirements," and TS 3.3.1 requirements for reactor trip system instrumentation.

b. Observations and Findings

On September 17, 1997, PT456 drifted up approximately 10 pounds per square inch gauge (psig) over an 8-hour period. On September 23, PT456 failed a channel check and was declared inoperable. The licensee initiated an LCO and placed Channel II in trip within 6 hours, per TS 3.3.1, Action 7. Occurrence Report (OR) 2-97-361 was generated to document the failure.

Work order #97006926 was issued for replacement of PT456. On September 24, the inspector observed the calibration and response time testing (RTT) of the replacement Barton Model 763 pressure transmitter. The calibration was performed in accordance with plant procedures with no discrepancies.

While the calibration and RTT were being performed, the licensee discovered that the replacement transmitter was not environmentally qualified (EQ) even though it had been issued as EQ. This was identified by Quality Control (QC) personnel while answering questions posed by maintenance concerning EQ splices for the transmitter pigtails. During the discussions, the QC supervisor recognized that the transmitter was not EQ, based on the purchase order number. Maintenance was immediately notified that the transmitter was not EQ and to suspend work. The licensee generated OR 2-97-369 to document the deficiency.

The licensee did not have any more Barton Model 763 transmitters in stock but was able to locate several at another plant and arranged to have two transmitters shipped. However, on October 2, due to complications with documentation and span differences, the licensee decided to install a Foxboro transmitter (Model NE11), the same as installed on Unit 1.

The Foxboro transmitter was installed under Design Change Package (DCP) S-97-2-9276. The physical changeout of the original Barton transmitter only required changing the mounting bracket and rerouting the sensing line. The inspectors reviewed the DCP and determined that it adequately addressed the mechanical aspects of the modification. Westinghouse Electric Corporation provided SNC an evaluation comparing the Foxboro and Barton transmitters which concluded that the performance of the two transmitters was the same with respect to the uncertainty calculations. However, the DCP and associated technical evaluation worksheet, did not address possible response ramp rate differences between the Foxboro and Barton transmitters.

The inspectors observed the calibration and RTT of the Foxboro transmitter. The calibration was completed again with no discrepancies. The transmitter response time was determined to be within the specified limit of 0.23 seconds.

Technicians identified a problem while installing the transmitter on the manufacturer-provided seismic mounting plate. The 3/8-inch Grade 5 bolts supplied with the mounting plate were too long. Therefore, the maintenance staff shortened and rethreaded the bolts. When the technicians attempted to torque one of the bolts, the threads stripped. The licensee initiated OR 2-97-378 to document the event. The licensee determined that the bolt failed due to poor workmanship when rethreading the bolt. The licensee replaced the provided Grade 5 bolts with Grade B7 bolts. The inspectors concluded that this was adequate.

An inspector accompanied licensee personnel during the at-power entry into Unit 2 Containment on October 3, 1997, to observe installation of the Foxboro transmitter. Inspectors attended the pre-job briefing and ALARA briefings for Radiation Work Permit (RWP) 2-97-2490. The briefings were comprehensive and complete. The containment entry was conducted in a professional manner. The entry team demonstrated teamwork and expeditiously completed the assigned tasks, in tight quarters and a hostile environment. However, one problem arose due to not energizing electrical outlets in the work area prior to the entry. The local portable air samples were unable to be collected. This pre-planning issue was discussed with the on-shift SS.

The inspectors subsequently reviewed the Foxboro and Barton RTT data, UFSAR Chapter 15, and background documentation. The inspectors identified an issue affecting the adequacy of the test procedure to accurately measure the sensor response time. Accurately measuring sensor response time is required to ensure that the total Reactor Trip System (RTS) response time is less than the 2 seconds assumed in the accident analysis. The RTS response time was defined in the background documentation and licensee submittal as "...the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage."

The TS currently requires periodic time response testing of the RTS from the output of the transmitters to the loss of stationary gripper coils. The response time portion for the sensor is measured prior to installing a new transmitter or after significant repairs to a transmitter. The current limit of 0.23 seconds for the transmitter was based on the slowest response time of a pressure transmitter determined through a historical record search of FNP transmitter response times.

While reviewing the response time traces, the inspectors noted that while the reference pressure transmitter was responding to the set ramp rate of approximately 540 psig/sec, the Foxboro was only capable of responding at approximately 300 psig/sec. This difference in response rate could be significant because the RTT procedure specified measuring the response time for only a 40 psig pressure drop whereas the actual pressure drop during an accident is about 400 psig from normal operating pressure (NOP) to the low pressure trip setpoint. If the transmitter was able to respond at the tested ramp rate the test could adequately measure the sensor response time. However, due to the transmitter's slower response rate and the small 40 psig pressure drop, current RTT does not accurately or conservatively measure sensor response time for high ramp rates.

The inspectors also reviewed the expected Reactor Coolant System (RCS) pressure ramp rate for large break Loss of Coolant Accidents (LOCAs) as identified in Section 15.4 of the UFSAR. Figures 15.4-3A through -3E indicated that core pressure drops to 1600 psig in less than 0.5 seconds. This ramp rate was much greater than even the tested 540 psig/second. The inspectors determined that, based on an actual response rate of 300 psig/second, the Foxboro transmitter could have a real response time of approximately 1 second for the expected transient. This is greater than the 0.23 seconds accounted for in the licensee's reactor protection system (RPS) response time equations.

This issue was discussed in detail with licensee management on October 8. On October 10, the licensee provided the inspectors with the test data packages for the most recent RPS and engineering safety features (ESF) response time testing. The test data showed that, even if 1 second was added to account for the slow response time of the Foxboro transmitter, the RPS and ESF time response would be within the required limits. Also, the licensee identified that the accident analysis for large break LOCAs did not depend on the control rod insertion to shutdown the reactor. The accident analysis determined that the reactor would be shutdown due to voiding and loss of moderator and would remain shutdown due to the injection of borated water from the Refueling Water Storage Tank (RWST) and cold leg accumulators. The accident analysis information along with the current test data alleviated the immediate safety significance of this issue.

c. Conclusions

Maintenance and support activities associated with the replacement of PT456 were generally well-controlled, and performed by competent and experienced personnel. However, a non-EQ instrument was inappropriately issued for an EQ application and a re-threaded bolt failed due to poor workmanship. Also, a technical issue concerning RTT and the capability of Foxboro pressure transmitters to respond to high ramp rates was identified. Due to the generic implications of this issue, further review will be conducted by the NRC. This is identified as IFI 50-364/97-11-01; RPS Response Time Testing.

M1.3 DCP S96-2-9060, Unit 2 R11/12 Containment Air Particulate and Gas Radiation Monitors (IP 62707 and IP 37551)

The DCP was performed under WO S00079791. The purpose of the DCP was to eliminate paper drive problems due to high flow rate through the paper drive unit. The DCP accomplished this by bypassing approximately 40% of the flow around the paper drive unit. The inspector reviewed the design calculations (SJ-95-1024-001, Rev.0) and verified that R11 still met the designed sensitivity with the reduced sample flow. The installation was performed in accordance with the DCP. Craftsmanship was good. The inspector verified that no new elbows or sharp bends were created which could affect the sample flow.

M8 Miscellaneous Maintenance Issues (IP 92700 and IP 92902)

M8.1 (Closed) Licensee Event Report (LER) 50-364/97-03; Failure To Perform Diesel Generator Surveillance Requirements Due To Procedural Inadequacy

The licensee determined that the required 18-month surveillance of TS 4.8.1.1.2.c.8 had not been performed on Unit 2 for EDG 1-2A. This issue was discussed in Section M1.8 of IR 50-348, 364/97-05 and was cited as an example of violation 50-348, 364/97-05-03.

The inspectors reviewed the Shared (Unit 1 and 2) Surveillance Schedule. This schedule had a note for EDG 1-2A that instructed the Operations group to ensure that the surveillance requirements of procedure FNP-0-STP-80.8, "Diesel Generator 1-2A 1000KW Load Rejection Test," Rev. 10, were accomplished for each Unit.

The inspectors were informed during discussions with licensee personnel that procedure FNP-0-STP-80.8 was to be replaced with surveillance test procedures FNP-1-STP-80.17 and FNP-2-STP-80.17. These procedures are currently in draft and will enhance the licensee's corrective actions by implementing unit-specific procedures for the 1-2A shared EDG. Similarly, procedure FNP-0-STP-80.9 for the 1C EDG was to be replaced with procedures FNP-1-STP-80.18 and FNP-2-STP-80-18.

Based upon the inspectors' review of documentation and the licensee's actions, this LER is closed.

M8.2 (Closed) LER 50-348/97-05; Failure To Perform Nuclear Instrumentation Surveillance Requirements Prior To Mode 2 And 3 Entry

The licensee discovered that the reactor trip instrumentation surveillance requirements of TS Table 4.3-1 were not met during a unit shutdown on March 15, 1997. This issue was discussed in Section M1.8 of IR 50-348, 364/97-05 and was an example of violation 50-348, 364/97-05-03.

The inspectors reviewed the new maintenance surveillance test procedures (Revision 0) for performing the nuclear instrumentation system (NIS) source range channel level trip calibration and functional test. The procedures are performed quarterly and ensure that the surveillance requirement is met for functional testing while in mode 1. The procedures are for source range channels N31 and N32 for both units.

The inspectors reviewed the licensee's Commitment Action Tracking Licensing Information Processing System (CATLIPS). This system tracked the licensee's corrective actions. Commitment #10172 of CATLIPS documents the commitment to change the UFSAR to allow performing NIS power range (PR) neutron flux low setpoint bistable calibration in mode 1 to ensure that TS 4.3.1.1 surveillance requirements are met for unit shutdowns.

The inspectors also reviewed the lessons learned training advisory notice associated with the issues discussed in LER 50-348/97-05. Based upon the inspectors' review of documentation and the licensee's actions, this LER is closed.

M8.3 (Closed) Violation (VIO) 50-348, 364/97-05-03; Failure To Follow Multiple TS Surveillance Requirements

The inspectors reviewed licensee corrective actions. Initial actions to alleviate the adverse conditions satisfactorily addressed the immediate issues. Licensee Event Report 50-348/97-05, Failure To Perform Nuclear Instrumentation Surveillance Requirements Prior To Mode 2 And 3 Entry, documented the failure to perform the quarterly functional tests and shiftly channel checks for the NIS source range (SR) and the NIS PR channels low flux trip for mode 2. This LER was discussed and closed in section M8.2 of this report. Licensee Event Report 50-364/97-03, Failure to Perform Diesel Generator Surveillance Requirements Due to Procedural Inadequacy, documented the missed surveillance associated with not conducting the 1-2A DG Load Rejection test for each individual unit prior to taking surveillance credit. This LER was discussed and closed in section M8.1 of this report. Selected updated Unit Operating Procedures (UOPs) were reviewed and determined to have been appropriately revised. Additionally, the Shared (Unit 1 and 2) Surveillance Schedule was reviewed and appropriate changes verified. The inspector verified that training was provided to appropriate personnel, concerning Code of Federal Regulations (CFR) 10 CFR 50.59 screening. Based on the licensee's actions, this VIO is closed.

M8.4 (Closed) IFI 50-348, 364/96-09-04: CCW HX Epoxy Coating and Broken Tubes

This item was opened to follow the performance of the epoxy coating process used on the Component Cooling Water (CCW) heat exchangers (HXs) and the adequacy of the licensee's Nonconformance Disposition Reports (NDR) in that the missing tube fragments would not impact the operability of the CCW system.

The inspectors reviewed the licensee's preventive maintenance (PM) tasks for the CCW HXs and verified that a task was added to specifically inspect the epoxy coating at 18-month intervals. The licensee inspected all three Unit 1 CCW HXs and one Unit 2 CCW HX after approximately three to four months of onservice time. Some minor deficiencies were identified and they were repaired prior to returning the HX to service. The inspectors examined two of the HXs when they were opened for the licensee's inspection. The coatings were intact with no signs of separation from the base metal or erosion of material.

To assess the adequacy of the licensee's NDRs, the inspectors reviewed LERs written from January 1, 1997 to September 15, 1997, for instances of foreign material problems in the CCW system. The inspectors also reviewed the documentation of licensee inspections of the CCW HXs and interviewed the licensee personnel who conducted the inspections. No issues involving foreign materials, i.e. tube fragments, were identified during the period reviewed. No further degradation of the tubes, tube failures and fragmentation, were identified by the licensee's subsequent inspections.

The inspectors concluded that the licensee's efforts to reduce erosion of the CCW HXs tube sheets through epoxy coating were effective and the coating was holding up well. The licensee's NDRs were accurate and thorough as demonstrated by no instances of tube fragments impacting the performance of the CCW system. The efforts to capture the broken and severed tubes and establish "fences" to prevent tube fragment migration were successful. Based on the inspectors review, this IFI is closed.

M8.5 (Closed) Inspector Followup Item (IFI) 50-348, 364/96-13-03: Foreign Material From Seal Injection System To Reactor Coolant Pump (RCP) Seals

During the past Unit 2 refueling outage (U2RF11), very small pieces of debris from the seal water injection filter O-rings were discovered in the downstream seal water supply check valves. The inspectors reviewed completed OR 2-96-325 and interviewed responsible personnel and management. As part of their corrective actions, licensee maintenance personnel inspected both seal injection filters, verified that existing O-rings were in place, and lubricated the O-rings per the vendor manual. All three seal injection lines were subsequently flushed, with no additional debris identified. Maintenance procedure FNP-0-MP-2.8, Replacement of Seal Injection Filters, Rev. 0, was written to ensure proper installation of seal water injection filters, including O-ring lubrication. A 10 CFR 50.59 safety evaluation was also documented regarding the potential introduction of foreign O-ring material into the reactor coolant pump seals. This evaluation concluded that, due to the size and constituency of the debris along with the torturous path of the RCP seal package, the likelihood of seal failure was minimal. This IFI is closed.

M8.6 (Closed) LER 50-364/96-03: Steam Generator Tube Degradation and Tube Status

This LER was provided to satisfy TS 4.4.6.5.c which requires that steam generator (SG) tube inspection results which fall into Category C-3 shall be considered a reportable event and reported pursuant to 10 CFR 50.73 prior to resumption of plant operation. The LER also served to satisfy TS 4.4.6.5.a which requires that following each In-Service Inspection (ISI) of SG tubes, the number of tubes plugged or repaired in each SG shall be reported to the Commission within fifteen days of the completion of the inspection, plugging, or repair effort. This LER is closed.

M8.7 (Closed) VIO 50-348, 364/97-130 01014: Failure To Prescribe Documented Instructions For Procedures to Implement Penetration Room Filtration (PRF) Testing and Operation

The inspectors reviewed the licensee's Commitment Action Tracking Licensing Information Processing System (CATLIPS), Reply to the Notice of Violation dated May 28, 1997, and procedures associated with the applicable corrective actions. Twenty procedures were reviewed and determined to have been appropriately revised in accordance with the corrective action plan. Based on this review of the corrective actions this VIO is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Unit 2 Rod Control Cluster Assemblies (RCCA) Full Withdrawn Rod Position Change

a. Review Scope (IP 37551)

On September 19, a resident inspector observed Operations, Instrument and Controls (I&C), and Engineering Support (ES) personnel implement FNP-2-ETP-3607, RCCA Fully Withdrawn Repositioning At Power, Revision 0, to fully withdraw the Unit 2 RCCAs from 225 steps to 226 steps.

b. Observations and Findings

This infrequently performed evolution was briefed in accordance with FNP-0-AP-92, Infrequently Performed Tests Or Evolutions, Revision 3, by the Unit 1 Operations Superintendent. The procedure was well written and controlled by the ES test director. Operations personnel implemented the procedure in a deliberate step-by-step manner under the direct supervision of the ES test director and oversight of the

Operations Superintendent. The evolution went smoothly except that annunciator FF5, COMP ALARM/ROD SEQ/DEV, came into alarm and would not clear. This condition was investigated and later explained to the inspector as an expected phenomenum when considering the pre-existing digital rod position indication (DRPI) Data A Channel failure of rod J09.

c. Conclusions

The evolution was properly controlled and the reason for the annunicator alarm was adequately understood.

E1.2 Generic Letter (GL) 95-05 Reportability and Safety Assessment

a. Review Scope (IP 37551)

By letter dated September 12, 1997, SNC addressed the reportability and safety assessment requirements of GL 95-05, Section 6. The Materials and Chemical Engineering Branch of the Office of Nuclear Reactor Regulation (NRR) reviewed SNC's letter using the criteria of GL 95-05, Section 6. Specifically, the NRC staff reviewed SNC's safety assessment, compensatory measures, and reportability determination (e.g., 10 CFR 50.72 or 50.73).

b. Observations and Findings

Voltage-based Steam Generator (SG) tube repair criteria was implemented at Units 1 and 2, in accordance with GL 95-05. SNC evaluated the affect that recent SG tube leak and burst test results have on the End-of-cycle (EOC) conditional leakage and probability of burst calculations, and concluded that inclusion of the latest leak and burst test results caused Units 1 and 2 to reach the NRC staff notification limits of GL 95-05.

GL 95-05 Section 6, Reporting Requirements, requires NRC staff notification under certain conditions. One condition that requires NRC staff notification occurs when a licensee determines that the EOC accident leakage will exceed the site-allowable leakage limit; another occurs when a licensee determines the EOC conditional burst probability exceeds 1×10^{-2}. SNC calculated a limiting probability of burst to be 1.4×10^{-3}, which is below the NRC staff notification level. However, when SNC incorporated the leak and burst test results from the recent Unit 1 and 2 tube pulls into the correlations used as part of the GL 95-05 leakage and probability of burst calculations, the projected EOC leakage from Unit 1 increased from 15.7 gpm to 20.4 gpm. This increase placed Unit 1 in the position of having exceeded the allowable leakage limit of 13.7 gpm. The revised EOC burst probability was calculated to be 1.2×10^{-2}. SNC also notified the staff that, with the most recent tube pull results in the leakage and burst correlations, the Unit 2 leakage was projected to exceed the allowable leakage limit on November 6, 1997. The probability of burst value remained under the NRC notification limit at 3.2×10^{-3}.

On September 12, 1997, SNC provided the NRC staff a "Generic Letter 95-05 Safety Assessment" for Unit 1 in accordance with the requirements of GL 95-05 Section 6. It was also part of a September 17, 1997, license amendment request for Units 1 and 2. The license amendment involved the reduction in the specific activity limits of dose equivalent I^{131} (DEI) steady state and transient values from 0.3 microcurie/gram to 0.15 microcurie/gram, and 18 micro curies/gram to 9 micro curies/gram, respectively. The DEI level reductions effectively increase the maximum allowable accident leakage limit associated with voltage-based repair criteria from 13.7 gpm to 23.8 gpm (room temperature conditions).

The staff reviewed the licensee's assessment against the criteria in Section 6 of GL 95-05. Specifically, the review included SNC's assessment of the safety significance, compensatory measures taken, and actions with respect to reportability of the event.

SNC's assessment of the safety significance of the increased EOC accident leakage was based on the actual plant steady state value of DEI (less than 0.01 microcurie/gram). Using actual plant conditions, SNC concluded that the radiological exposure from SG tube leakage in the event of a main steam line break (MSLB) would not have exceeded the licensing basis. However, the licensee failed to explicitly address the radiological consequences of a MSLB assessed in two ways: (1) assuming a preexisting iodine spike and (2) assuming an accident-initiated iodine spike. Since the licensee implicitly addressed both cases when the licensee changed its administrative limits for both steady state and transient values of DEI, the staff concurred with SNC's safety assessment with respect to leakage. Regarding the increased burst probability, SNC cited operator action and engineering judgement to conclude the 1.2×10^{-2} burst probability was not safety-significant. The staff concurred with SNC's conclusion.

SNC evaluated the reportability of the revised leakage and burst probability numbers and determined the reportability of the issue was covered by the requirements of GL 95-05 and no other reportability requirements (e.g., 50.72 or 50.73) applied. The staff reviewed the reporting requirements and concluded that SNC has complied with the requirements of Technical Specification 3.4.6 by having followed the applicable reporting requirements outlined in Section 6 of GL 95-05.

c. Conclusions

The NRC staff found SNC's "Generic Letter 95-05 Safety Assessment" in response to the GL 95-05 requirements to be adequate. With respect to leakage, the actual plant conditions combined with the administrative limits established by SNC appear to ensure EOC accident leakage will not result in radiological exposures exceeding regulatory limits. With respect to conditional burst probability, the staff concludes the small increase is not safety-significant. The licensee's compensatory actions were appropriate, and the reporting requirements appear to have been adequately addressed.

E1.3 SWS Discharge Pipe Corrosion (IP 37551)

In response to inspector concerns regarding excessive SWS pipe corrosion (see report section O2.1), the licensee initiated WOs #S97006667 and #S97006668 to clean the Unit 1 and 2 SWS pipe and conduct nondestructive examinations. Under the cognizance of ES engineers, plant personnel performed ultrasonic testing (UT) of the affected SWS piping. The UT results, including pit depth measurements, were then transmitted to corporate engineering, Southern Company Services (SCS), for review as a Request for Engineering Assistance (REA) 97-1557. In particular, SCS was requested to calculate the minimum acceptable wall thickness. Inspectors observed evidence of the SWS pipe cleaning, UT, and subsequent priming of the external piping surface. An inspector also reviewed the UT data recorded in the WOs, and reviewed the SCS reply to REA 97-1557. The average UT pipe wall thickness readings were typically 0.51 inches, with a wall thickness of as low as 0.390 inches in the corroded areas. As noted in the original piping specification, this piping was purchased for a minimum thickness of 0.428 inches. The licensee concluded that the deepest pit was approximately 0.125 inches deep. Subsequent calculations by SCS concluded the minimum allowed wall thickness was 0.229 inches. Consequently, the existing SWS pipe condition was acceptable. Overall, licensee response to the identified SWS pipe corroded conditions was prompt and effective.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Open) IFI 50-348, 364/97-10-02: UFSAR Reverification Corrective Actions

a. Inspection Scope

The inspectors selected 18 of the 868 UFSAR discrepancies for followup. During this report period item #089 concerning the capacity of the EDG air start systems was reviewed. An inspector reviewed the documented response to the discrepancy, licensee calculations, UFSAR Sections 8.3 (Onsite Power Systems) and 9.5.6 (Diesel Generator Starting System), NRC Standard Review Plan (SRP) 9.5.6, Rev. 1, and pre-startup EDG test data. The review of the calculations was limited to the air starting requirements for the Colt-Pielstic PC2 EDGs.

b. Observations and Findings

The concern as stated in the UFSAR Verification database was: "The statement that the accumulators have the capacity for five air starts should be investigated to its origin in order to establish if it is a design requirement which has been satisfied or if it is a one-time or periodic testing requirement."

The licensee's closeout response to the question was that this was a design requirement and that based on calculations "no explicit testing of this requirement and revision to the UFSAR" was necessary. The inspector requested copies of the design calculations SM-90-1779-001, "Diesel Generator - Air Start System," Rev. 1, and SM-90-1779-02, "Air Start System Leakage Rate," Rev. 1, for review. These calculations were performed in 1992 and 1993.

The inspector reviewed the above calculations, including the licensee assumptions. The inspector also performed independent calculations using the licensee's assumptions and formulas, but with the actual pressure drop observed by the inspector during routine EDG surveillance starts, to validate the licensee's methodology. The inspector also compared the licensee's results to original EDG startup test data. The inspector determined the licensee's calculational methodology was non-conservative. The licensee had failed to compare their methodology against data from original EDG startup tests, specifically performed to validate the design of the air start system receiver capacity, nor did the licensee's calculations reflect actual EDG surveillance data. The inspector concluded that the licensee's response to the UFSAR discrepancy lacked thoroughness in that the licensee's review failed to recognize the existence of actual startup test data or use pressure drops observed during routine surveillance testing.

The inspector also concluded that the startup test data demonstrated that the EDGs were capable of five sequential starts from one receiver without recharge. In response to the inspector's comments, the licensee revised their resolution of item #089 to document existence of the startup tests.

While researching the above issues the inspectors identified another UFSAR discrepancy. UFSAR section 8.3.1.1.7.2, Response to Design Basis Events, states that the maximum required loads will not exceed the continuous rating of any of the four design basis diesel generators. This statement was not accurate in that current design basis load for the 1C EDG exceeds the continuous rating but is less than the 2000 hour rating. The inspectors have reviewed the loading of the 1C EDG previously and determined that exceeding the continuous rating was acceptable because the licensee's TS surveillance tests the 1C EDG to the 2000 hour rating. This UFSAR discrepancy was identified to the licensee for correction.

c. Conclusions

The licensee resolution of UFSAR discrepancy #089 was not thorough or complete. Calculations supporting the design of the air start system were non-conservative and were not validated against existing test data. However, the inspector verified that startup test data demonstrated the air start system was adequately sized.

This IFI remains open pending additional review of the UFSAR reverification corrective actions.

E8.2 (Closed) IFI 50-348, 364/96-13-07: Certain High Energy Line Break (HELB) Isolation Sensors Not Described In UFSAR

This NRC identified UFSAR discrepancy was originally entered into CATLIPS for tracking as commitment #10253. However, as part of the upcoming conversion to the Improved Standard TS (ISTS), TS 3.3.3.7 for "High Energy Line Break Isolation Sensors" is to be removed and relocated to the UFSAR. Consequently, the licensee has closed CATLIPS commitment #10253 and opened item #506 in the ISTS conversion action item database to track this UFSAR discrepancy. An inspector reviewed item #506 of the ISTS conversion database. Since this item will be included in the ISTS review, this IFI is considered closed.

E8.3 (Closed) VIO 50-364/96-155-01014: Steam Generator Tube Flaws With F* Distance

Closeout of this VIO was previously documented in Section M8.1 of NRC IR 50-348, 364/96-09.

E8.4 (Closed) Unresolved Item 50-348(364)/97-201-01: Unprotected CST Connections

On October 21, 1994, during the safety system self-assessment (SSSA) of the auxiliary feedwater (AFW) system, the licensee discovered that a number of piping and transmitter tubing connections to the Unit 1 and Unit 2 Condensate Storage Tanks were not provided with missile protection as described in the UFSAR. Section 9.2.6.6 of the UFSAR stated that the lower 12 feet of the CST was designed to withstand any rupture caused by missiles. To resolve this issue, the licensee issued Incident Report 1-94-299, Licensee Event Report (LER) 94-005-00, and an UFSAR change and associated 10 CFR 50.59 Safety Evaluation. The inspectors noted that the licensee's LER (94-005-00) and the subject Safety Evaluation had been reviewed previously by NRC as documented by NRC Reports 95-20 and 96-07, respectively.

The licensee's corrective action involved issuing a change to the UFSAR to reflect the as-built configuration of the CST piping without the tornado missile protection. The 10 CFR 50.59 Safety Evaluation of the proposed UFSAR change was completed on November 17, 1994. The 10 CFR 50.59 Safety Evaluation included a question (question number 6) asking, "May the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR?" The licensee answered this question "No." However, the inspectors noted that as a result of this condition, a tornado missile could damage some CST connections, thereby resulting in a loss of inventory from the safety-related tank and affecting the operability of the entire AFW system.

A probablistic risk analysis was prepared and documented in Calculation REES-F-94-014 which indicated that the impact frequency with which a tornado missile could strike exposed CST piping was on the order of 1.0×10^{-8} per year. The safety evaluation for the UFSAR change compared the calculated impact frequency with which a tornado missile could strike exposed CST piping (approximately 1.0×10^{-8} per year) to the probability of occurrence of a design basis external event (approximately 1×10^{-7} per year) and concluded that this postulated tornado event was not required to be analyzed as an "accident" in the UFSAR. The UFSAR was subsequently revised to include the PRA results and to delete the requirement for missile protection of the subject CST connections. However, the inspectors concluded that the comparison was not an appropriate justification to determine that an unreviewed safety question (USQ) did not exist when the as-built plant configuration did not conform to the configuration described in the UFSAR.

In a letter to NRC dated July 11, 1997, the licensee committed to provide missile protection for the subject connections by March 15, 1998. The inspector found that the licensee had developed and issued Design Change Packages (DCPs) 97-1-9172 and 97-2-9173 to add tornado missile protective structures to CST connections in the lower 12' of the tank. The CST connections identified in the DCP to be tornado missile protected were tank drain, vacuum degasifier tank connection, and level transmitters with associated electrical conduit.

10 CFR 50.59 allows licensees to make changes to the facility as described in the safety analysis report, without prior Commission approval, unless the proposed change, involves a change in the technical specifications or an unreviewed safety question. The NRC has reviewed the circumstances related to this issue and determined that a USQ did exist; however, as described in the cover letter to this report, the NRC is exercising enforcement discretion to not cite the violation in accordance with Section VII.B.6 of the Enforcement Policy. This unresolved item is closed.

E8.5 (Closed) URI 348,364/97-201-02: Tornado Protection of CST Level Instrumentation

UFSAR Sections 3.2.1.5 and 9.2.6.1 and Table 3.2-1 state that the AFW system instrument and control (I&C) system equipment and CST equipment were classified as Category I, respectively. UFSAR Section 3.5.4 states that Category I equipment and piping outside containment are either housed in Category I structures or buried underground. However, during the walkdown of the Unit 1 CST, the inspectors observed that the safety-related CST level transmitters and enclosures, as well as the associated cables and conduits, were outside, and routed above ground around the tank perimeter without missile protection.

In a letter to NRC dated July 11, 1997, the licensee committed to provide missile protection for the CST level transmitters and associated conduits by March 15, 1998. As stated earlier, the licensee had issued DCPs 97-1-9172 and 97-2-9173 to install missile protection at both the Unit 1 and Unit 2 CSTs. The inspector verified that these instruments were included in the scope of the DCPs.

This unresolved item is dispositioned with URI 97-201-01, as described in E8.4, and is now closed.

E8.6 (Closed) URI 50-348,364/97 201-03: AFW Check Valve Reverse Flow Testing

The inspectors identified that TDAFW pump discharge check valves V003 or V002D, F, and H were not included in the IST program for a reverse flow valve closure test as required by the ASME Code. The licensee agreed that either check valve V003 or check valves V002D, F and H were required to close in order to perform the required safety function. The licensee issued OR 1-97-048 on March 3, 1997, to assure that required corrective actions were implemented in a timely manner. The licensee revised the Unit 1 and 2 IST Plans, FNP-1-M-46 and FNP-2-M-071 to require reverse flow testing of TDAFW check valves V002D, F and H every refueling outage. The IST Program was also revised to identify that these valves have a safety function in the open and closed position.

Surveillance Test Procedure FNP-1-STP-22 29, "Turbine Driven Auxiliary Feedwater Check Valve Reverse Flow Closure Operability Test," Revision 0, dated April 18, 1997, was issued to implement reverse flow testing on the subject valves for Unit 1. The procedure was implemented on Unit 1 and satisfactorily completed on May 21, 1997.

In a letter to NRC dated July 11, 1997, the licensee indicated that all corrective actions had been completed including issuance of procedures to perform the required testing. This response was in error in that the Unit 2 procedure was not issued until September 12, 1997. The inspector found that the scheduled date of completion for issuance of the Unit 2 Surveillance Procedure was in accordance with the corrective action described in Occurrence Report 1-97-048 and the Open Commitment Tracking Report dated September 15, 1997. In accordance with these documents, the Unit 2 Surveillance Test Procedure FNP-2-STP-22.29 was scheduled to be issued and completed satisfactorily prior to Unit 2 Startup from refueling outage RF12 which is scheduled for Spring 1998. The licensee informed the inspector of this discrepancy and indicated that a revised submittal would be provided if necessary to clarify the procedure status. The inspector concluded that based on review of the Occurrence Report and the Commitment Tracking database that the corrective actions were being properly tracked and completed and no additional response on this item would be required.

The inspector concluded that the failure to reverse flow test either TDAFW check valve V003 or check valves V002D, F and H as required by the ASME Code was a violation of TS Section 4.0.5 which requires inservice testing of ASME Code classes 1, 2, and 3 pumps and valves in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This is identified as Violation 50-348,364/97-11-02, Failure to Perform Adequate IST of TDAFW Check Valves on Cessation or Reversal of Flow.

The unresolved item is closed.

E8.7 <u>(Closed) URI 348,364/97-201-04: AFW Check Valve Forward Flow Testing</u>

The inspectors reviewed Surveillance Test Procedure FNP-1-STP-22.13, "Turbine Driven Auxiliary Feedwater Pump Check Valves Flow Verification," Revision 14, dated May 7, 1996. The inspectors noted that because of the testing lineup with the minimum recirculation flow path open the flow through TDAFW check valve V003 would be on the order of 530 gpm when the pump was operated at 625 gpm. The team concluded that the acceptance criteria of 625 gpm in Section 2.2 of FNP-1(2)-STP-22.13 was not consistent with the actual test flow. The licensee agreed with the finding and issued Temporary Change Notice (TCN) 14A and 12A to revise procedures FNP-1(2)-STP-22.13.

The licensee's correspondence dated July 11, 1997, stated that procedures have been revised to reflect current acceptance criteria and a review of other procedures was in progress. The inspector reviewed Surveillance Test Procedures FNP-1(2)-STP-22.13 dated February 28, 1997, and verified that they had been properly revised to reflect the correct acceptance criteria for full flow testing of check valves Q1N23V003 and Q2N23V003. The acceptance criteria for total flow through the valve was changed to 450 gpm and the flow was being measured down stream of the minimum recirculation flow path at flow indicator FI-3229.

The failure to have adequate acceptance criteria in the surveillance test procedure for verifying the forward flow for check valve V003 was a violation of 10 CFR 50 Appendix B, Criterion V. This is identified as Violation 50-348,364/97-11-03, TDAFW Battery Installation and Check Valve Test Deficiencies.

Based on the above, the unresolved item is closed.

E8.8 <u>(Closed) URI 50-348,364/97-201-05: TDAFW Pump Battery Testing</u>

The inspectors questioned the lack of service testing for the TDAFW UPS Batteries to demonstrate the ability of the battery to meet the design duty cycle specified in the battery Design Basis Calculation 07597-E-106. The inspectors did not have any immediate safety concern, since the licensee's maintenance and testing provided reasonable assurance that the battery could support the assigned load. The inspector followed up on this item and concluded based on the review that a failure to have a test program and procedures for service testing of the TDAFW Class 1E battery to ensure that the battery would meet the required duty cycle was a violation of 10 CFR 50 Appendix B, Criterion XI and TS Section 6.8.1.a. This is identified as Violation 50-348,364/97-11-04, Failure to Implement a Test Program for Service Testing of the TDAFW Battery.

The licensee committed to perform battery service testing during refueling outage RF14 for Unit 1 and RF12 for Unit 2 and to establish a task to perform a service test every 18 months thereafter. The architect engineer provided the licensee a draft procedure and safety evaluation in Letter No. FP 97-0179 dated April 4, 1997. The Procedure FNP-1-EMP-1352.04, "Turbine Driven Auxiliary Feedwater (TDAFW) UPS Battery Service Test," Revision 0 was issued on April 22, 1997. The procedure was satisfactorily completed on Unit 1 on April 23, 1997. The inspector reviewed the test results and the procedure and found both to be acceptable. The battery test was a combined test to demonstrate that the as-found battery capacity was adequate to supply calculated design basis accident load requirements for 2 hours and station blackout load requirements for 4 hours. The load profile consisted of 70 amperes for the first minute followed by 53 amperes for the following 239 minutes. The acceptance criteria was that the battery terminal voltage remained greater than 42.6 volts dc after being subjected to the service discharge test profile above.

This unresolved item is closed.

E8.9 (Closed) URI 50-348,364/97-201-06: TDAFW Battery Installation

The inspectors reviewed the Unit 2 TDAFW battery installation and found that various structural and electrical components were not installed in accordance with the manufacturer's drawings and instructions. The inspectors identified 6 specific deficiencies and 5 of those deficiencies involved discrepancies between the battery installation and drawings or procedures. The following deficiencies were noted:

- Five polystyrene spacers were installed between the battery cell and the end rail where none were required.
- Structural steel bracing in the rear of the rack did not agree with the drawing.
- Bolts were missing from the upper- and lower-tier tie rod brackets.
- Silicon bronze bolting hardware was utilized at the cable terminations in lieu of stainless steel hardware.
- The intercell battery connections were torqued to 75 in-lbs instead of the required 125 in-lbs specified in the battery manufacturer's instruction manual.
- The battery rack steel rails and tie rods exhibited corrosion.

The inspectors' review of these issues concluded that the failure to install the Unit 2 TDAFW battery and rack in accordance with drawings, procedures or instructions was a violation of 10 CFR 50 Appendix B, Criterion V. This is identified as an example of Violation 50-348,364/97-11-03, TDAFW Battery Installation and Check Valve Test Deficiencies.

In licensee correspondence dated July 11, 1997, the licensee indicated that the battery rack would be rebuilt per approved drawings and the work would be completed by June 15, 1998.

The licensee issued REA 97-1408 to reconcile the differences between the battery rack design and the installed configuration; and REA 97-1444 to revise TDAFW battery maintenance procedures and appropriate documentation to clarify acceptable fastener material. The licensee received a response to REA 97-1408 in a letter dated July 31, 1997, and indicated that the rack support frames had been installed 29 inches apart instead of 25 inches apart as required by the drawings. The recommended corrective action was to disassemble the rack and relocate one of the support frames to within 25 inches of the other in accordance with the design drawing. The response also included a proposed work sequence to disassemble and reassemble the battery rack. The licensee received a response to REA 97-1444 in Letter No. FP 97-0396 dated July 31, 1997. Attached to this letter was a draft ABN which provided information reflecting the acceptability of silicon bronze or stainless steel fastener material for use on the Units 1 and 2 TDAFW UPS battery intercell and field cable connections.

The licensee found that the maintenance procedures (for individual cell replacement as well as periodic cleaning and inspection) specified torque values that were less than recommended by the vendor. The licensee evaluated this issue and determined that the torque values did not affect the battery's safety function based on frequent periodic battery maintenance and information received from the vendor that indicated that acceptable connection resistance readings were obtainable over a wide range of torque values. The licensee indicated a review of torque values for all safety-related batteries was in progress and the procedures would be revised appropriately. The licensee indicated the specific TDAFW battery procedures would be revised by June 15, 1998.

The licensee issued Deficiency Report #537766 to cleanup the battery rack corrosion and install missing hardware. This work was completed on February 12, 1997.

The unresolved item is closed.

E8.10 <u>(Closed) IFI 50-348/97-201-07: CST Level Alarm</u>

The inspectors reviewed Calculation SM-87-4380-001, Revision 0 and identified the following two concerns:

- The drift error for the sensor was not addressed in the calculation.

- The total instrument tolerance calculated did not include the deadband of 1% of span. (The inspectors found that it was not clear from the design guidance document as to the circumstances when to use deadband in uncertainty calculations.)

The inspector found that Calculation SM-87-1-4380-001 had been superseded by Calculation SJ-97-1407-001 and Calculation SM-97-1407-002. Calculation SM-97-1407-002, Condensate Storage Tank Low-Low Level Alarm Setpoint, Revision 0, dated August 18, 1997, determined the lowest allowable low-low level alarm setpoint on the Unit 1 and 2 CSTs. The lowest allowable low-low level alarm setpoint was determined to be 1.456 feet from the bottom of the tank. However, to ensure adequate margin, the licensee administratively set the low level alarm setpoint at 5 feet - 3 inches or 63 inches. The administrative setpoint of 63 inches was used as an input into Calculation SJ-97-1407-001, Calculation to Establish the Total Loop Uncertainty for Loops L-515 and L-516, Revision 0, dated August 19, 1997. This calculation determined the total loop tolerance for L-515 and L-516 and applied the loop tolerance to the designated setpoint and process limit to verify that all inaccuracies and allowances made would not cause the alarm initiation to fall outside safe process limits. The inspector reviewed portions of these calculations and verified that sensor and rack drift had been adequately addressed.

In regard to the issue on deadband, the licensee had revised the Project Desk Instruction (PDI) 005.16, Process Instrumentation and Control Setpoints, dated August 26, 1997, to clarify when the deadband component should be used in establishing acceptable setpoint tolerances in uncertainty calculations. The inspector found the revision to PDI 005.16 clarifying when deadband should be considered to be acceptable.

This item is closed.

E8.11 (Open) URI 50-348,364/97-201-08: Tornado Protection of TDAFW Pump Vent Stack

The inspectors observed that the safety-related TDAFW pump vent stack was installed on the roof of the Auxiliary Building and was not protected from tornado generated missiles. UFSAR Section 3.5.4 states that Category I equipment and piping outside containment are either housed in Category I structures or buried underground. UFSAR Sections 6.5.1, 3.2.1.3, 3.2.1.5 and Table 3.2-1 state that AFW system equipment and piping are Category I. The inspectors noted that the requirements of Criterion III and V of Appendix B to 10 CFR Part 50 were not met because the installed condition of the TDAFW pump vent stack did not conform to FNP design and licensing basis. This issue will remain open pending additional NRC review.

E8.12 (Open) URI 50-348,364/97-201-09: Tornado Missile Spectra

The inspectors observed that the safety-related emergency diesel generators and the station blackout diesel generator exhaust silencers for both units were installed on the roof of the diesel generator building. The equipment was judged to be protected against horizontal generated missiles by the building walls. However, a concern was identified that the equipment was susceptible to vertical missiles and other non-horizontal missiles. The licensee took the position that the design basis for FNP was for horizontal missiles only. The issue was left as an unresolved item pending further review by NRC to determine if the tornado missile protection in the FNP design and licensing bases included missile spectra other than horizontal missiles. In a letter to NRC dated May 28, 1997, the licensee provided additional information to support its position on vertical missiles. This information was being reviewed by NRR and the review is scheduled to be completed by November 1997. This item will remain open pending completion of this review.

E8.13 (Closed) IFI 50-348/97-201-10: CST Level Transmitter Freeze Protection

The inspectors identified three potential deficiencies with the installation of the CST level transmitter Q1P11LT516 and associated freeze protection. In response to these concerns, the licensee issued Work Orders (WO) # 97001089, 97002478, and 97003706 to have maintenance inspect and evaluate the adequacy of the heat tracing and insulation for level instruments Q1P11LT515 and LT516 and repair as required. All work orders were completed on May 21, 1997. The inspector reviewed the work history records to determine if the work was performed satisfactorily and addressed the concerns. Based on this review, this item is closed.

E8.14 (Closed) URI 50-348,364/97-201-11: AFW UFSAR Discrepancies

The inspectors identified the following discrepancies between the UFSAR, as-built plant and design:

1. UFSAR Section 6.5.2.2.4 stated, "All valves in the AFW flow path from the condensate storage tank to the steam generators were normally open, with the exception of the fail open AFW flow control valves." This statement implied the AFW flow control valves were not normally open. In accordance with the plant's operating procedures, these valves were normally maintained in the open position. The inspectors considered the UFSAR statement was not correct.

2. UFSAR Section 3K.4.1.2.7, item F, states the lowest safety-related equipment in the main steam room is the atmospheric relief valves at elevation 133 ft. 3 inches. During the design inspection audit in February - March 1997, a walkdown identified that the lowest safety-related components in the main steam valve room were the Motor Driven Auxiliary Feedwater (MDAFW) and Turbine Driven Auxiliary Feedwater (TDAFW) discharge valves (HV-3227A, B, C; HV-3228A, B, C). The lowest solenoid valve was located at elevation 131' - 0". Because these valves were located at a lower elevation than the atmospheric relief valves, as stated in the above UFSAR statement, a concern existed regarding the validity of the statement in the UFSAR that the plant personnel would have approximately 4 additional hours to isolate the turbine driven pump discharge through the feedwater line break before water levels in the main steam room could potentially approach the bottom of a safety-related component. In addition, the inspectors observed that the limit switches associated with the main steam isolation valves (MSIVs) were located less than 1 foot above the floor, which was also below the analyzed flood level.

3. DCP S-96-1-9008-0-001 replaced two - 3 amp fuses on the output of TDAFW pump UPS rectified output with 5 amp fuses. However, UFSAR section 8.3.3.2.C on page 8.3-41 still had references to 3 amp fuses.

In response to items 1 and 2, the licensee issued ABN 97-0-1074 to revise the UFSAR. Design Calculation BM-97-1074-001, "Basis for time in UFSAR Section 3K.4.1.2.7, Item F " was developed to support ABN 97-0-1074 and document the design basis for the UFSAR statement regarding time available to isolate the TDAFW pump flow through a feedwater line break in the main steam valve room. The calculation results indicated that the plant personnel would have approximately 3.18 hours instead of 4 hours to isolate the turbine driven pump discharge from the feedwater line break before the water level in the main steam room reaches the bottom of the safety related solenoid located at elevation 131'.

UFSAR Section 6.5.2.2.4 was being revised to identify that the normal position of the auxiliary feedwater control valves was open except during auxiliary feedwater system testing. In addition, Section 6.5.2.3.3 was being revised to reflect the normal position of the AFW flow control valves and their operation on an automatic pump start signal.

The inspector found that the issue on fuse sizing had been previously addressed by the licensee in ABN 93-0-0224, Revision 2, which was issued in letter No. FP 95-0511 from Southern Company Services to Southern Nuclear dated August 31, 1995. This ABN (93-0-0224) included a revision to UFSAR Section 8.3.3.2.C to delete the fuse rating from the UFSAR on the basis that fuse design was controlled by the fuse manuals (Documents A-181987 and A-201987) and therefore, fuse rating details need not be documented in the UFSAR. Attached to the ABN was a 50.59 Safety Evaluation, and UFSAR markups. The inspector reviewed the UFSAR markup of the proposed change and found it to be acceptable. The ABN was approved by Plant Operations Review Committee (PORC) on February 8, 1996. Based on these actions, item 3 is being dispositioned in accordance with Enforcement Guidance Memorandum (EGM) 96-005 with no further action.

The inspector concluded that deficiencies 1 and 2 were examples of a violation of 10 CFR 50.71(e). In accordance with the Enforcement Policy, the failure to update the UFSAR normally would be categorized as a Severity Level IV violation. However, as discussed in the cover letter of this inspection report, enforcement discretion is being exercised in accordance with Section VII.B.3 of the Enforcement Policy.

The unresolved item is closed.

E8.15 <u>(Open) URI 50-348,364/97-201-12: Stress Analysis Temperature</u>

This issue was identified to review the licensee's root cause evaluation and corrective actions for the design deficiency documented in Deficiency Notice (DN) 97-001 involving incorrect dimensions and an incorrect temperature utilized in the Component Cooling Water (CCW) piping stress calculations. The subject DN was generated to document the as-found condition and initiate the corrective action process. The DN stated that a Root Cause Evaluation Team (RCET) made up of SCS, SNC, and BPC representation would be established to determine the root cause; perform a review; and provide corrective action recommendations.

The inspector found that this root cause evaluation was still in progress. However, the corrective action to revise the Unit 1 and 2 CCW stress calculations had been completed. The licensee had issued REA 97-1415 to revise the Unit 1 and 2 CCW Stress Calculations. A list of the individual calculations as well as the criteria used to determine if a calculation required revision were documented in DN 97-001. Table A of DN 97-001 listed the CCW stress calculations for Units 1 and 2. The DN indicated in the notes whether a Change Notice had been issued against a calculation or if the calculation was required to be revised. Licensee letter No. FP 97-0394 dated July 31, 1997 which was a final response from the design for REA 97-1415, indicated that 30 CCW Stress calculations required revision as a result of this deficiency. However, it indicated that additional calculations were revised as part of the snubber reduction program for a total of 35 calculations that were revised.

This item will remain open pending review of the results of the licensee's root cause evaluation and proposed corrective actions.

E8.16 <u>(Closed) URI 50-348,364/97-11-13: MOV Design Basis Differential Pressure</u>

The inspectors identified that the design basis differential pressures identified in MOV Design Basis Document, U418109, Revision A, for CCW system containment isolation valves MOV 3046, 3052, and 3182 were non-conservative as the effect of post-LOCA containment pressure was not considered. These valves were located in the piping at penetration numbers 42 and 14, which were the CCW supply to the RCPs and the CCW return from the RCP oil coolers. UFSAR Table 6.2-31 identified these penetrations as Type II. UFSAR Section 6.2.4.1 defined Type II penetrations as serving those lines that connect directly to the containment atmosphere. Therefore, the design-basis differential pressure for the containment isolation valves should have considered the maximum post-LOCA containment pressure that could exist when the valves operate. The inspector concluded that the licensee's failure to check and verify the adequacy of the design was a violation of 10 CFR 50 Appendix B, Criterion III. This is identified as an example of Violation 50-348,364/97-11-05, Design Control Measures Did Not Ensure that Calculations Were Verified and Controlled Adequately.

ABN 97-0-1098 was issued by the licensee on September 9, 1997, to revise Units 1 and 2 MOV Design Basis Drawings, U418109 (Sheets 76B, 78B, and 91B) and U418110 (Sheets 61B, 63B, and 76B), to incorporate the new closing differential pressure values for the subject MOVs. The ABN and associated 50.59 Safety Evaluation was reviewed and found to be acceptable. The new closing differential pressure value for MOVs 3046 and 3182 was 2[illegible] psid and 52 psid for MOV 3052.

The unresolved item is closed.

E8.17 <u>(Closed) IFI 50-348,364/97-[illegible]C1-14, CCW Pump Testing</u>

Bechtel Calculation (41.4) BM-95-0776-001, "CCW System Evaluation Using Degraded CCW Pump Curve," Revision 0, was performed under REA 95-0776. This calculation documented the acceptability of the CCW system performance with the pumps degraded approximately 10% from the vendor test curves and stated that this degraded pump curve will be used for comparing test data to verify the performance capability of the CCW pumps. The inspectors noted that the Inservice testing of the CCW pumps was performed in accordance with procedure FNP-2-STP-23.1, "2A Component Cooling Water Pump Quarterly Inservice Test," Revision 14, and similar procedures existed for the other two pumps. The inspectors also noted that the CCW Functional System Description (FSD) and procedure had not been revised to incorporate the results of the above calculation. An IFI was identified to review the revised IST procedure and associated safety evaluation.

The inspector found that ABN 97-0-1080, Revision 0 and associated 10 CFR 50.59 Safety Evaluation provided for revising the UFSAR, CCW FSD, SWS FSD, RHR FSD and CCW P&IDs to allow CCW pump operation with the minimum flow line isolated. This ABN revised affected documentation based on using minimum analyzed CCW pump performance data for various normal and accident configurations with the minimum flow line open and for operating the CCW pumps with the minimum flow line isolated. The inspector reviewed the Component Cooling Water Functional System Description, (A-181000), Revision 8 and observed that Revision 0 of ABN 97-0-1080 was incorporated into the FSD on July 8, 1997. The inspector reviewed the 10 CFR 50.59 Safety Evaluation for ABN 97-0-1080, Revision 1 and no unreviewed safety question was identified. This item is closed.

E8.18 <u>(Closed) URI 50-348,364/97-201-15: Post Modification Testing</u>

This issue involved Design Change Package (DCP) 96-0-9012-2-006, "Process Coating for CCW Heat Exchangers," which provided direction for modification of the CCW heat exchangers by application of an epoxy coating (Plastocor) to the tubesheets, channel head, channel cover, channel head shell relief line, approximately 12 inches into the service water inlet and outlet lines, and 12 inches of the inlet end of the tubes. The DCP, along with REA 96-1211, also provided direction for plugging and stabilizing tubes. Procedure FNP-0-ETP-4418, "CCW Heat Exchanger Epoxy Coating Application," Revision 1, implemented the epoxy coating and Work Orders 96001476, 96001477, and 96001478 installed the stabilizing rods in the tubes. The inspectors noted that neither the procedure used to apply the epoxy coating nor DCP 96-0-9012-2-006 required post-modification testing to ensure design flow capability had been maintained. The inspector concluded that the concern that a flow test was required was based on engineering judgement. The inspector could not identify any specific regulatory requirements that would require flow testing. Therefore, this issue is closed.

In a letter to NRC dated July 11, 1997, the licensee indicated that appropriate procedures would be revised by August 15, 1997, to delineate post-mod testing requirements for maintenance replacement design changes. The inspector reviewed Plant Modifications Procedure FNP-0-PMP-100, "Design Change Engineering Evaluation Preparation," Revision 16 and noted that it had been revised to delineate requirements for post-modification testing for Maintenance Replacement DCPs.

E8.19 (Open) IFI 50-348,364/97-201-16: Calculation Discrepancies

The inspectors identified several discrepancies in Calculation SC-96-1211-002, "CCW Heat Exchanger Maintenance Repairs," Revision 1. This calculation documented the seismic and mechanical acceptability of the modification. The deficiencies were noted in Sections 2.1, 2.3, 2.4, and 2.5 of the calculation. The licensee issued REA 97-1407 to revise the calculation. This item was identified to review the revised calculation. The inspector followed up on this item and found that the calculation had not been revised. However, in a letter to NRC dated July 11, 1997, the licensee committed to have the calculation revised by October 15, 1997. This item will remain open pending issuance and review of the revised calculation.

E8.20 (Closed) URI 50-348,364/97-201-17: Drawing and Procedure Discrepancies

Procedures FNP-2-SOP-23.0A, "Component Cooling Water System," Revision 5; FNP-2-SOP-2.1A, "Chemical and Volume Control System," Revision 8; and FNP-2-SOP-1.1A, "Reactor Coolant System," Revision 6, were checklists for the normal positions of valves and circuit breakers. The inspectors identified numerous differences between the P&IDs for the system (D-205002 Sheet 1, Revision 21; Sheet 2, Revision 10; and Sheet 3, Revision 2) and procedures FNP-2-SOP-23.0A and FNP-2-SOP-2.1A concerning the existence of caps on vent and drain lines. The inspectors noted that item 5 of safety system self-assessment (SSSA) observation CCW-CM-01 was related to this item but was apparently not corrected by the licensee. The SSSA observation was issued on April 19, 1990. The inspectors reviewed this issue and concluded that the failure by the licensee to take corrective action for an identified deficiency was a violation of 10 CFR 50 Appendix B, Criterion XVI. This is identified as Violation 50-348,364/97-11-06, Failure to Take Corrective Action for Difference Between CCW System Piping and Instrument Drawings and System Operating Procedures.

Based on this action the unresolved item is closed.

E8.21 (Closed) URI 50-348,364/97-201-18: CCW UFSAR Discrepancies

The inspectors identified the following discrepancies in the UFSAR:

1. Table 9.4-6A listed the room temperature for the Component Cooling Pump Room at the beginning of the post-accident period as 119 degrees F, whereas Table 3.11-1 indicated a design temperature of 104 degrees F for the same room.

2. There are four relief valves (Q2P17V153, V154, V155, V158) on CCW piping between the inboard and outboard containment isolation valves. These relief valves represent a release path to the environment. However, these relief valves were not listed in UFSAR Table 6.2-39 as containment isolation valves.

3. Table 9.3-1 did not include valve HV 2229, which was also a safety-related, air-operated valve that received a safety injection actuation signal (SIAS).

4. Several differences existed between UFSAR Tables 9.2-6 and 9.2-7 and Tables T-1 through T-5 in the CCW FSD. For example, UFSAR Table 9.2-6 listed the charging pump lube oil cooler flow as 20 gpm and FSD Table T-2 lists this flow as 30 gpm.

With the exception of item 1 above, the other deficiencies are considered violations of 10 CFR 50.71(e) for failure to ensure that the latest developed material was included in the UFSAR. However, as discussed in E8.14, enforcement discretion is being exercised regarding these violations. In regard to item 1, the inspector reviewed this item and concluded that no deficiency existed.

This unresolved item is closed.

E8.22 <u>(Closed) URI 50-348,364/97-201-19: TS Change for Auxiliary Building Battery</u>

Unit 1 and 2 Technical Specifications Section 4.8.2.3.2.c.5 for Auxiliary Building battery service test specified a minimum cell voltage requirement of 1.75 volts dc. Surveillance Procedures FNP-1-STP-905.1 and FNP-2-STP-905.1, which perform the required service test on the batteries specified a minimum acceptable voltage at the end of the service test which was higher than that specified in the TS. The inspectors noted that the 10 CFR 50.59 Safety Evaluations performed for PCN B-92-0-8099 and the changes to FNP-1(2)-STP-905.1 stated that TS were not affected. However, these changes required battery terminal voltages higher than those specified in TS.

10 CFR 50.36(c)(3), Technical Specification, Surveillance Requirements, states that surveillance requirements are related to test to assure that the necessary quality of systems and components are maintained and that the limiting conditions for operation will be met. The failure by the licensee to identify a required TS change and to submit the application for license amendment is identified as Violation 50-348,364/97-11-07, Failure to Change TS for Auxiliary Building Battery.

In a letter to NRC dated July 11, 1997, the licensee committed to submit a revised TS by December 31, 1997.

Based on this action the unresolved item is closed.

E8.23 <u>(Closed) URI 50-348,364/97-201-20: Fire Barrier Penetration Seal Documentation</u>

The inspectors noted that silicone foam fire penetration seal 45-121-26 contained copper tubing. This configuration deviated from the tested configuration, and an engineering evaluation of the acceptability of the deviation had not been documented in accordance with UFSAR Section 9B.2.2.5.3. Inspection Report 97-12 identified other concerns with the as-built configurations of silicone foam fire barrier penetration seals. An inspector followup item was identified to review the licensee's evaluations of deviations from tested fire barrier configurations. This issue is added as another example to be reviewed as part of IFI 50-348,364/97-12-01, Review of Engineering Evaluations to Establish the Fire Rating or Fire Resistant Capabilities of Fire Rated Silicone Foam Penetration Seals. Therefore, the unresolved item is closed.

E8.24 (Closed) URI 50-348,364/97-201-21: Electrical UFSAR Discrepancies

The inspectors identified the following discrepancies in the UFSAR:

1. UFSAR Section 8.3.1.1.3.A.2 stated that the unit auxiliary transformer "B" megavolt-ampere (MVA) rating at 65 degrees C was 47.99 instead of 46.7 as shown on drawing D-202700.

2. Section 8.3.1.1.9B referred to Section 8.3.1.1.3 for interrupting capacities for distribution panels. However, Section 8.3.1.1.3 did not include interrupting capacity data for distribution panels.

3. Section 8.3.1.2 stated there were 21 600-V/208-V motor control centers, however, the actual number of motor control centers identified in the UFSAR totaled 19.

In regard to items 1 and 2 above, the inspector concluded that these were additional violations of 10 CFR 50.71(e). However, as discussed in E8.14, enforcement discretion is being exercised regarding these violations. The inspector noted that item 3 had previously been identified by the licensee's UFSAR Verification Program as Item #070M. A 50.59 safety evaluation, FVP-025 (B19500 Section 8), had been prepared to revise the UFSAR. Included as part of the UFSAR change was a markup of the UFSAR deleting the reference to the quantity of load centers. The inspector reviewed the Safety Evaluation and UFSAR Markup and found them to be acceptable. The inspector considered the licensee's corrective action for item 3 to be adequate.

The unresolved item is closed.

E8.25 (Closed) URI 50-348,364/97-201-22: Control of Calculations

The inspectors identified that in several cases calculations that had previously been superseded were not identified as such on the calculation index; design basis calculations were not appropriately revised to show the existing design condition; and affected calculations were not revised when new calculations were performed. The inspector followed up on this issue and concluded that the licensee's design control measures did not ensure that calculations were verified and controlled adequately. The failure to ensure adequate design controls for calculations is identified as an example of Violation 50-348,364/97-11-05, Design Control Measures Did Not Ensure that Calculations Were Verified and Controlled Adequately.

Based on the above, the unresolved item is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls (IP 71750)

R1.1 Partial Entries Into Contaminated Areas

An inspector observed a number of partial entries into contaminated areas. In general, maintenance personnel were more conscientious than operators in applying protective actions to prevent the inadvertent spread of contamination during partial entries. The inspector observed some examples of poor operator practices during partial entries. These observations were discussed with Operations management.

R2 Status of Radiological Protection and Chemistry Controls Facilities and Equipment (IP 71750)

R2.1 Radiologically Controlled Area, Units 1 and 2

During tours of the radiologically controlled areas (RCA) of the auxiliary building for Units 1, inspectors observed that overall cleanliness and housekeeping was good. Ongoing decontamination efforts by the Health Physics (HP) department to reduce contaminated surface areas continue to be successful. Floor spaces in the RHR pump rooms, and certain spent fuel pool (SFP) cooling pump skids, have been decontaminated due to HP's aggressive efforts. In concert with decontamination efforts, HP has also redesigned catch devices to minimize contamination and still control minor leaks.

R8 Miscellaneous RP&C (IP 92904)

R8.1 (Closed) IFI 50-348, 364/97-10-03: Review Licensee Evaluation for Extended Onsite Storage of Contaminated Wet Resin

The licensee performed inspections of all the Surepaks on September 9 and 19, 1997. The inspector observed the pre-job brief and portions of the licensee's inspections of the Surepaks and steel liners containing the contaminated wet resin. The inspections were thorough and concentrated on the outer surface of the liners which were raised by a crane for the inspection. The surface of the liners were acceptable with only minor surface corrosion visible. The licensee also obtained water samples of the water in the liner and the standing water in the bottom of the Surepak for pH analysis. This analysis indicated that the water was of neutral pH and was not accelerating the minor corrosion observed. The licensee performed followup inspections on October 16 and determined that the resins were not generating any measurable quantities of gaseous products. The inspector reviewed the licensee's procedure and schedule for periodically inspecting the Surepak. The inspection data sheet required an inspection on a quarterly basis. The licensee considers that these inspections will be adequate to identify liner degradation before it becomes a problem. This item is closed based on the licensee's actions.

P1 Conduct of EP Activities (IP 71750)

P1.1 Emergency Plan Drill

On September 10, 1997, resident inspectors and the NRR project manager participated in an announced drill of the licensee's emergency plan. As drill players, the inspectors considered the drill scenario reasonably challenging. The Technical Support Center (TSC), Operations Support Center (OSC) and Emergency Operations Facility (EOF) were all manned and fully operational in a timely manner. During the drill, emergency response personnel properly characterized evolving events and made accurate and timely emergency classifications and notifications.

S1 Conduct of Security and Safeguards Activities (71750)

S1.1 Routine Observations of Plant Security Measures

During routine inspection activities, inspectors verified that portions of site security program plans were being properly implemented. This was generally evidenced by: proper display of picture badges by plant personnel; appropriate key carding of vital area doors; adequate stationing/tours in the protected area by security personnel; proper searching of packages/personnel at the primary access point and service water intake structure; and adequate condition of security systems. Security personnel activities observed during the inspection period were performed acceptably. Site security systems remained functionally adequate to ensure physical protection of the plant.

S3 Security and Safeguards Procedures and Documentation (IP 71750)

S3.1 Safeguards Material In The MCR Not Positively Controlled

On September 17, 1997, a resident inspector reviewed the following safeguards documents located in the Unit 1 Shift Supervisor's (SSs) desk drawer in the at-the-controls (ATC) area of the Main Control Room (MCR): a) Security Plan, Revision 32; b) Contingency Plan, Revision 7; c) Contingency Implementing Procedures, and d) Security Procedures. The inspector verified that all these safeguards plans and procedures were of the latest revision. However, the inspector identified the following problems:

a) The folders containing the Contingency Implementing Procedures and Security Procedures were not marked as Safeguards Information (SGI);

b) Access to the Unit 1 SS's desk was not positively controlled by a lock nor constantly attended by the SS; and,

c) Although the MCR is an access-controlled vital area, access to the MCR is not limited solely to those personnel authorized to review SGI. Personnel not authorized to review SGI were regularly granted access to the MCR, including the ATC area.

The inspectors met with the Deputy Security Chief, and then with an Operations Superintendent, to express concern that SGI located in the MCR was not properly controlled to prevent access by unauthorized personnel. All SGI was promptly removed from the MCR, placed in the Central Alarm Station (CAS) which was required to be continually staffed by security personnel, and properly marked as SGI. Inspectors verified the licensee's corrective actions.

FNP-0-AP-72, Protection of Safeguards Information, Revision 9, Step 6.2.1, states "SGI is required to be under the control of an authorized individual while in use to prevent unauthorized disclosure to persons without a need to know. The requirement for control of SGI is met if the matter is attended by an authorized individual even though the information is not constantly being used." SGI in the MCR was not being attended by an authorized individual during those periods every day when the Unit 1 SS leaves his desk, and especially when both SSs leave the ATC area. Step 9.1 of AP-72 also requires each document that contains SGI to be positively marked in a precise manner, that was not apparent on the SGI maintained in the MCR.

The provisions of AP-72 were consistent with the requirements of 10 CFR 73.21(d) for storing SGI in a locked security storage container whenever left unattended; and, 10 CFR 73.21(e) for marking SGI in a conspicuous manner as "Safeguards Information." Failure to adequately control and mark the SGI maintained in the ATC of the MCR constituted a violation of NRC regulations and licensee procedures as identified as VIO 50-348, 364/97-11-8, Unattended And Unmarked SGI Left in the MCR. However, licensee corrective actions have been prompt and effective to ensure SGI was controlled and marked pursuant to regulatory requirements.

F8 Miscellaneous Fire Protection Issues (IP 92904)

F8.1 (Closed) IFI 50-348, 364/96-006-07: Fire Main Failures

This item was opened pending metallurgical analysis of the failed piping and implementation of longterm corrective actions. Southern Company Services provided the results of the metallurgical analysis and recommendations for action via letter dated December 5, 1996. The inspectors previously reviewed this issue, but were unable to close the item because the recommended corrective actions had not been implemented.

The licensee implemented the recommended corrective action on September 1, 1997. Licensee staff identified all outside fire protection piping and inspected it to verify the integrity of the insulation and flashing and that no water had penetrated and soaked the insulation. No discrepancies were identified. The licensee also implemented an eighteen month preventive maintenance task to perform this inspection. The inspector verified the new PM task was entered into the information management system.

The inspectors concluded that the corrective actions were thorough. Based on the licensee's corrective action, this IFI is closed.

V. Management Meetings and Other Areas

X1 Review of Updated Final Safety Analysis Report Commitments

A recent discovery of a licensee operating its facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters, except for:

1) EDGs running in excess of their continuous rating (see Section E8.1); and,

2) R-29 not being routinely source checked (see Section O2.1).

X2 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on October 21, 1997, after the end of the inspection period. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Harris, SNC Corporate Engineering Manager*
M. Ajluni, SNC Corporate Licensing Manager*
R. Badham, Safety Audit and Engineering Review (SAER) Supervisor
R. Coleman, Maintenance Manager
C. Collins, Operations Superintendent - Administration
S. Fulmer, Technical Manager
D. Gambrell, Design Team Leader, Southern Company Services (SCS)*
D. Grissette, Operations Manager
P. Harlos, Plant Health Physicist
R. Hill, General Manager
C. Hillman, Security Chief
R. Johnson, Operations Superintendent - Support
D. Jones, Configuration Management Manager
H. Mahan, SNC Corporate Senior Engineer*
R. Martin, Maintenance Team Leader
D. McKinney, Engineering and Licensing Manager*
M. Mitchell, Health Physics Superintendent
R. Monk, Engineering Support Supervisor
D. Morey, Vice President - Farley Nuclear Project*
C. Nesbitt, Assistant General Manager, Plant Support
R. Ponder, SNC Corporate Senior Engineer*

D. Shelton, SCS Engineering Manager*
M. Stinson, Assistant General Manager, Operations
G. Wilson, SCS Senior Engineer*

NRC

J. Zimmerman, NRR Project Manager

* Supported NRC inspection at SNC Corporate offices and attended pre-exit interview on September 19, 1997.

INSPECTION PROCEDURES (IP) USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls In Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of LERs
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
IFI	50-348, 364/97-11-01	Open	RPS Response Time Testing (Section M1.2).
VIO	50-348, 364/97-11-02	Open	Failure to Perform Adequate IST of TDFW Check Valves on Cessation or Reversal of Flow (Section E8.6).
VIO	50-348, 364/97-11-03	Open	TDAFW Battery Installation and Check Valve Test Deficiencies (Sections E8.7 and E8.9).
VIO	50-348, 364/97-11-04	Open	Failure to Implement a Test Program for Service Testing of the TDAFW Battery (Section E8.8).
VIO	50-348, 364/97-11-05	Open	Design Control Measures did not ensure that calculations were verified and controlled (Sections E8.16 and E8.25).

VIO	50-348, 364/97-11-05	Open	Inadequate Corrective Action To Resolve Differences Between CCW System P&IDs And Operating Procedures (Section E8.20).
VIO	50-348, 364/97-11-07	Open	Auxiliary Building Battery Surveillance Test Criteria Inconsistent With TS (Section E8.22).
VIO	50-348, 364/97-11-08	Open	Unattended And Unmarked SGI Left in the MCR (Section S3.1).

Closed

Type	Item Number	Status	Description and Reference
IFI	50-348, 364/97-10-03	Closed	Review Licensee Evaluation for Extended Onsite Storage of Contaminated Wet Resin (Section R8.1).
VIO	50-348, 364/97-11-11	Closed	Unattended And Unmarked SGI Left In The MCR (Section S1.1).
LER	50-364/97-03	Closed	Failure To Perform Diesel Generator Surveillance Requirements Due To Procedural Inadequacy (Section M8.1).
LER	50-348/97-05	Closed	Failure To Perform Nuclear Instrumentation Surveillance Requirements Prior To Mode 2 And 3 Entry (Section M8.2).
VIO	50-348, 364/97-05-03	Closed	Failure To Follow Multiple TS Surveillance Requirements (Section M8.3).
IFI	50-348, 364/96-09-04	Closed	CCW HX Epoxy Coating and Broken Tubes (Section M8.4).
IFI	50-348, 364/96-06-07	Closed	Fire Main Failures (Section F8.1).
IFI	50-348, 364/96-13-03	Closed	Foreign Material From Seal Injection System To RCP Seals (Section M8.5).
IFI	50-348, 364/96-13-07	Closed	Certain HELB Isolation Sensors Not Described In UFSAR (Section E8.2).
LER	50-364/96-03	Closed	Steam Generator Tube Degradation and Tube Status (M8.6).

VIO	50-364/96-155-01014	Closed	Steam Generator Tube Flaws With F* Distance (Section E8.3).
VIO	50-348/97-130-01014 50-364/97-130-01014	Closed	Failure To Prescribe Documented Instructions For Procedures To Implement PRF Testing and Operation (Section M8.7).
URI	50-348, 364/97-201-01	Closed	Unprotected CST Connections (Section E8.4).
URI	50-348, 364/97-201-02	Closed	Tornado Protection of CST Level Instrumentation (Section E8.5).
URI	50-348, 364/97-201-03	Closed	AFW Check Valve Reverse Flow Testing (Section E8.6).
URI	50-348, 364/97-201-04	Closed	AFW Check Valve Forward Flow Testing (Section E8.7).
URI	50-348, 364/97-201-05	Closed	TDAFW Battery Testing (Section E8.8).
URI	50-348, 364/97-201-06	Closed	TDAFW Battery Installation (Section E8.9).
IFI	50-348/97-201-07	Closed	CST Level Alarm (Section E8.10).
IFI	50-348/97-201-10	Closed	CST Level Transmitter Freeze Protection (Section E8.13).
URI	50-348, 364/97-201-11	Closed	AFW UFSAR Discrepancies (Section E8.14).
URI	50-348, 364/97-201-13	Closed	MOV Design Basis Differential Pressure (Section E8.16).
IFI	50-348, 364/97-201-14	Closed	CCW Pump Testing (Section E8.17).
URI	50-348, 364/97-201-15	Closed	Post Modification Testing (Section E8.18).
URI	50-348, 364/97-201-17	Closed	Drawing and Procedure Discrepancies (Section E8.20).
URI	50-348, 364/97-201-18	Closed	CCW UFSAR Discrepancies (Section E8.21).
URI	50-348, 364/97-201-19	Closed	TS Change for Auxiliary Building Battery (Section E8.22).
URI	50-348, 364/97-201-20	Closed	Fire Barrier Penetration Seal Documentation (Section E8.23).

URI	50-348, 364/97-201-21	Closed	Electrical UFSAR Discrepancies (Section E8.24).
URI	50-348, 364/97-201-22	Closed	Control of Calculations (Section E8.25).

Discussed

Type	Item Number	Status	Description and Reference
IFI	50-348, 364/97-10-02	Open	UFSAR Reverification Corrective Actions (Section E8.1).
URI	50-348, 364/97-201-08	Open	Tornado Protection of TDAFW Pump Vent Stack (Section E8.11).
URI	50-348, 364/97-201-09	Open	Tornado Missile Spectra (Section E8.12).
URI	50-348, 364/97-201-12	Open	Stress Analysis Temperature (Section E8.15).
IFI	50-348, 364/97-201-16	Open	Calculation Discrepancies (Section E8.19).