U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-348 and 50-364
License Nos.:	NPF-2 and NPF-8
Report Nos.:	50-348/98-05 and 50-364/98-05
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Farley Nuclear Plant, Units 1 and 2
Location:	7388 N. State Highway 95 Columbia, AL 36319
Dates:	July 12 - August 29, 1998
Inspectors:	T. P. Johnson, Senior Resident Inspector J. H. Bartley, Resident Inspector R. K. Caldwell, Resident Inspector C. W. Rapp, Senior Project Engineer J. J. Blake, Reactor Inspector (SNC Corporate Office)
Approved by:	Pierce H. Skinner, Chief Reactor Projects Branch 2 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

FARLEY NUCLEAR POWER PLANT UNITS 1 and 2

Nuclear Regulatory Commission Inspection Report 50-348,364/98-05

This integrated inspection to assure public health and safety included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a six week period of onsite inspection by the resident inspectors.

Operations

- Operator response to a Unit 1 circulating water pump trip was prompt, and demonstrated good training and plant awareness (Section 01.2).

- The licensee appropriately and conservatively responded to steam generator 1B primary to secondary leakage including enhanced training and plant procedure revisions. The Unit 1 shutdown was well controlled and coordinated (Section 01.3).

- Procedures needed for mid-loop operations contained adequate information and appropriate detail to satisfy the concerns expressed in Generic Letter 88-17. All required reactor vessel level indications were functioning properly and closely monitored by the operators (Section 01.4).

- The licensee response to the component cooling water heat exchanger leak was timely and well coordinated. Shift supervisor command and control of the event response was good (Section 01.5).

- Strike contingency plans were detailed, met regulatory requirements, and demonstrated a proactive approach to a possible labor action (Section 06.1).

Maintenance

- Observed maintenance and surveillance testing activities were satisfactorily performed (Sections M1.1).

Engineering

- Licensee plans for inspection and replacement of Reactor Vessel Baffle-Former bolts were comprehensive. (Section E2.2)

- The licensee's organization and administration of steam generator replacement preparations for Farley Units 1 and 2 were comprehensive and on schedule. (Section E6.1)

- The licensee's resolution of Update Final Safety Analysis Report (UFSAR) discrepancies was typically thorough and correct. However, the 10 CFR 50.59 review of one UFSAR discrepancy package was inadequate and failed to identify an Unreviewed Safety Question (USQ). Also the inspectors identified two additional UFSAR discrepancies that were overlooked by the licensee (Section E8.3)

Plant Support

- Routine tours of the plant's radiological areas, including the Unit 1 containment, demonstrated good health physics oversight (Section R2.1).

- An emergency preparedness drill was appropriately conducted and was a good training exercise (Section P1.1).

- Security facility walkdown and routine observations demonstrated appropriate security personnel and equipment performance (Sections S1.1 and S2.1).

- The Shift Supervisor's command and control in response to a smoke alarm and report of smoke in Document Control was good (Section F1.1).

REPORT DETAILS

Summary of Plant Status

At the beginning of this reporting period, Unit 1 was operating at or near full power and had been on line since starting up from the last refueling outage on June 3, 1997. On July 19, 1998, the unit was reduced to 75% power due to a circulating water pump trip. Full power was re-achieved on July 23. On August 17, the unit was shutdown and cooled down to Mode 5 to locate and repair a steam generator tube leak.

At the beginning of this reporting period, Unit 2 was operating at or near full power and had been on line since starting up from the last refueling outage on May 17, 1998. The unit operated at or near full power during the period.

I. Operations

O1 Conduct of Operations

O1.1 Routine Observations of Control Room Operations (71707 and 40500)

The inspectors observed that control room professionalism and communications remained good. Operating crew demeanor, team work and conduct were professional and effective. Operator attentiveness to Main Control Board (MCB) annunciator alarms and responses to changing plant conditions were prompt. The operating crew consistently demonstrated a high level of awareness of existing plant conditions and ongoing plant activities, including during the Unit 1 forced outage.

The inspectors routinely reviewed the Technical Specification (TS) Limiting Conditions for Operation (LCO) tracking sheets. All tracking sheets for Units 1 and 2 reviewed by the inspectors were consistent with plant conditions and TS requirements.

O1.2 Unit 1B Circulating Water Pump Trip (71707)

On July 19, the 1B circulating water pump tripped. As required by the abnormal operating procedures (AOPs), operators reduced the unit to 75% power. Electrical checks on the motor were satisfactory and the pump was restarted; however, a subsequent trip occurred. The licensee determined that a grounded electrical power cable was the cause. The cable was replaced and the unit was returned to full power on July 23. The licensee intends to check and replace similar cables during the next scheduled unit refueling outages.

The inspectors concluded that operations and maintenance personnel reacted properly to this condition. Control room operator response was prompt and demonstrated good training and plant awareness.

01.3 Unit 1 B Steam Generator (SG) Primary to Secondary Leak

a. Inspection Scope (71707)

The inspectors reviewed the licensee's actions regarding a primary to secondary leak on the 1B SG.

b. Observations and Findings

The 1B SG has had minor primary-to-secondary leakage for several cycles. The leak rate was determined to be stable at about 4 gallons per day (gpd), following the unit restart from the Spring 1997 outage. The licensee also discussed this leak with Westinghouse and EPRI representatives. The licensee initially believed that the leak was from a tube plug or sleeve, but did not rule out other possibilities. About July 1, the licensee observed N-16 spikes on the "B" steam line radiation monitor. The licensee performed the following additional monitoring activities on the 1B SG:

- Isotopic analysis once per day
- Tritium analysis once per day
- Gaseous activity once every 12 hours
- Sodium activity once every 12 hours

The licensee confirmed that the spikes were valid with these other analyses. Initially, the spikes were less than 25 gpd and the leak rate then returned to approximately 4 gpd. However, in early August the leak rate increased to approximately 65 gpd and by mid August the leak rate increased to 85 gpd. On August 15, the licensee made a conservative decision to shutdown the unit, and repair the SG leak. The unit was shutdown on August 17 for a 16 day forced outage. After the shutdown and 1B SG primary side entry, leakage was observed from tube R25C51.

The licensee's Technical Specification (TS) and AOP administrative limit was 140 gpd. An industry suggested limit of greater than a 60 gpd change per hour for a sustained period was incorporated into procedures. The licensee used 20 minutes as the sustained period definition. If this leakage value was exceeded, the licensee was to commence a unit shutdown within one hour.

As a precaution, prior to the unit shutdown the licensee conducted simulator crew training for SG tube leak and rupture actions. The inspectors attended four of these sessions. Additionally, crew briefings occurred and night orders addressing the SG leakage issues were provided. The training required implementation of AOP and EOP actions.

The inspector observed that the Unit 1 shutdown was well controlled and coordinated by the shift supervisor (SS). The inspector reviewed the official copy of the shutdown procedure and found that the SS was maintaining procedure signoffs up-to-date. Minor equipment problems which occurred during the shutdown were responded to appropriately.

c. Conclusion

The inspectors concluded the licensee appropriately and conservatively responded to this SG leakage problem including enhanced training and plant procedure revisions.

O1.4 Unit 1 Mid-loop Operations

a. Inspection Scope (71707)

The inspectors observed licensee preparations for establishing mid-loop conditions on Unit 1 in accordance with FNP-1-UOP-4.3, Mid-loop Operations, Revision (Rev.) 5. The inspectors also performed MCR observations during mid-loop conditions.

b. Observations and Findings

The inspectors observed MCR operations during mid-loop conditions established and maintained from August 21-22, and August 28-29, due to the high core decay heat level. The inspectors observed the detailed pre-evolution briefings which covered initial conditions, pre-cautions, and industry events. The inspectors reviewed several procedures that were needed for mid-loop operation and found they contained adequate information and appropriate detail to satisfy the concerns expressed in Generic Letter 88-17. All required reactor vessel level indications were functioning properly and closely monitored by the operators.

c. Conclusions

The inspectors concluded that the licensee adequately prepared for and satisfactorily conducted Unit 1 mid-loop operations.

O1.5 Component Cooling Water (CCW) Heat Exchanger (HX) Tube Rupture (71707)

On August 19, a tube in the 1C CCW HX ruptured. An inspector was in the MCR and observed the licensee's response. The inspector concluded that the licensee's response was timely and well-coordinated. The Shift Supervisor's command and control of the event response was good.

O2 Operational Status of Facilities and Equipment

O2.1 General Tours of Safety-Related Areas (71707)

The inspectors observed the physical condition of plant equipment and structures, and verified that safety systems were properly maintained and aligned. No significant issues were identified.

O2.2 Inspections of Safety Systems (71707)

Inspectors verified the operability of selected safety systems. These systems were verified to be properly aligned and maintained. The Unit 1 Auxiliary Feedwater System (AFW) system was observed as having several check valves leaking (V002D, V003, and V002C). The licensee was appropriately monitoring AFW system temperatures on a periodic basis. The inspector verified that these leaking check valves were being addressed in the upcoming refueling outage. The Unit 1 High Head Safety Injection System (HHSI) flow element (FE) 943 had observed wet and dry boric acid deposits. The licensee had previously identified and evaluated this leakage to ensure the leakage was within auxiliary building leakrate requirements and for bolting integrity.

O6 Operations Organization and Administration

O6.1 Strike Contingency Plans (92709)

The licensee's contract with union personnel expired during the report period. The licensee established a business continuation plan in case of a union labor action. The plan included staffing, organization, support, and administrative related contingencies.

The inspectors reviewed the plan; verified the qualifications and actions for the TS required staffing of operations, the control room, the fire brigade, and other support organizations; discussed the plans' implementation with management; and, reviewed site access contingencies. The inspectors concluded that the licensee's plan was detailed, met regulatory requirements, and was workable. The licensee demonstrated a proactive approach to a possible labor action.

O8 Miscellaneous Operations Issues (90712, 92700)

O8.1 (Closed) Licensee Event Report (LER) 50-348/97-06; Steam Generator Tube Degradation and Tube Status
(Closed) LER 50-364/98-03; Steam Generator Tube Degradation and Tube Status

Licensee plans to replace Unit 1 SGs in the spring of the year 2000, and Unit 2 SGs in the spring of 2001. The resident and specialist inspectors reviewed these SG activities during the last two outages, and these two LERs. No safety concerns were identified.

O8.2 (Closed) LER 50-348/97-05-01; TS Surveillances

This event was initially documented in NRC Inspection Report (IR) 50-348, 364/97-05 and the failure to meet a TS required surveillance was identified as an example of VIO 97-05-03, Failure to Follow Multiple TS Surveillance Requirements. Revision 0 of the LER was closed in IR 97-11. Revision 1 of the LER was issued to document the same event occurring on Unit 2. The inspectors reviewed the updated LER and determined no new issues were identified.

08.3 (Closed) LER 50-364/98-01: Manual Reactor Trip Due to Dropped Control Rod K-2
(Closed) LER 50-364/98-01-01: Manual Reactor Trip Due to Dropped Control Rod K-2 and F-10

These events were discussed in NRC IR 50-346, 364/98-03. The root cause of the dropped rods was unable to be determined. However, the fuses appeared to have failed due to fuse fatigue. Fuses for Unit 1 and Unit 2 Rod Control moveable gripper and stationary circuits have been replaced. The licensee was conducting additional evaluations to determine the appropriate application and change frequency of these fuses. No new issues were revealed by these LERs.

08.4 (Closed) LER 50-348/98-01: Inadequately Performed Surveillance Due To Improper Calculation of Average Disintegration Energy (E-Bar)

The licensee reported that since September 1986, the Reactor Coolant System (RCS) specific activity (E-Bar) surveillance had been inadequately performed resulting in both Units operating in a condition prohibited by TS. When the procedure for determining E-Bar was revised, the TS limits were miscalculated. Since this revision, all calculated E-Bar values have been lower than actual. The licensee recalculated the TS limits and determined the current Unit 1 and Unit 2 specific activities were within the corrected values. Additionally, the licensee did a bounding analysis on the historical data and compared the highest recorded specific activity to the lowest corrected TS limit value and determined the TS limits were never exceeded. The licensee also updated the current procedure to ensure the proper determination of E-Bar. This failure constitutes a violation of minor significance and is not subject to formal enforcement action.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments (61726 and 62707)

The inspectors witnessed or reviewed portions of selected maintenance, surveillance, and test activities. For those maintenance and surveillance activities observed or reviewed, the inspectors determined that the observed activities were conducted in a satisfactory manner and that the work was properly performed in accordance with approved maintenance work orders. The inspectors also determined that the observed activities were performed in a satisfactory manner and met the requirements of the technical specifications.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.2 Reactor Vessel (RV) Baffle-Barrel-Bolting Program

a. Inspection Scope (37550)

The inspectors reviewed the licensee's plans for inspection and replacement of RV baffle-former bolting during the Fall Unit 1 refueling outage.

b. Observations and Findings

The inspectors met with SCS Nuclear Maintenance Support personnel to review the plans for inspection and replacement of baffle-former bolts during the Fall 1998, Unit 1 Refueling Outage. The inspectors also received a briefing on the remote bolt inspection and replacement tool that was being fabricated.

The reviewed outage plans called for the core barrel to be removed from the RV and placed in the refueling canal. The remote tool will be used to do 100% inspection of the baffle-former bolts and a replacement of about 600 bolts. The plans included a schedule of about five days for ultrasonic inspection of the bolts and 21 days for bolt replacement activities.

c. Conclusions

Licensee plans for inspection and replacement of Reactor Vessel Baffle-Former bolts were comprehensive.

E6 Engineering Organization and Administration

E6.1 Steam Generator Replacement (SGR) Organization

a. Inspection Scope (50001)

Farley Unit 1 SGR has been scheduled for March 2000, and Farley Unit 2 SGR has been scheduled for March 2001. The inspectors reviewed licensee and contractor specifications, agreements, procedures, and correspondence pertaining to these SGRs.

b. Observations and Findings

The Southern Nuclear Operating Company, Farley Project, established a project management organization for SGR as a direct report to the vice president of the Farley project. The licensee contracted with Westinghouse to provide Model 54F replacement steam generators (RSGs) as essentially "like-for-like" replacements for the Model 51 original steam generators (OSGs).

The SGR responsibilities were shared by the staffs of an Engineering Manager, an Installation Manager, a Quality Manager, and a Project Controls Manager all reporting to the SGR Project Manager. Additional responsibilities in the areas of health physics and operational impacts (procedures and operator training) were being assigned to appropriate representatives of the Farley plant staff. The inspectors reviewed the responsibilities and activities of the Engineering, Installation, and Quality Managers including various contracts, agreements, procedures, and schedules. Discussions, and review of schedules, correspondence, and progress reports, indicated that the engineering activities by the various groups were being well coordinated by the Engineering Manager including any required TS changes or revised UFSAR accident analyses. Identification and ordering of required plant, support, and consumable materials appeared to be on schedule. Review of detail periodic quality activity reports indicated an active surveillance program by the licensee's quality inspector. The licensee was preparing to send an additional quality inspector to monitor RSG assembly due to increased fabrication activity.

c. Conclusions

The licensee's organization and administration of steam generator replacement preparations for Farley Units 1 and 2 was comprehensive and on schedule.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Licensee Event Report (LER) 50-348, 364/97-09: Lack of Missile Protection For Service Water Flow Switches (90712, 92700)

This LER has been discussed in NRC IR 97-07 and 98-02. The licensee evaluated the potential flooding problem due to line cracks and determined that they are not licensed to consider moderate energy line breaks in seismically supported moderate energy line (which includes the Seismic Category-I service water piping in the diesel generator building). The inspectors have reviewed the response and concluded the licensee's reasoning was adequate.

The inspector concluded that this was a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. Consistent with Section VII.B.1 of the NRC Enforcement Policy, this licensee-identified and corrected violation is being treated as a Non-Cited Violation (NCV) and is identified as NCV 50-348, 364/98-05-01, Inadequate Missile Protection For Service Water Flow Switches.

E8.2 <u>(Closed) Violation 50-348, 364/97-14-05, Failure To Provide Tornado Missile Protection for Turbine Driven Auxiliary Feedwater (TDAFW) Pump Vent Stack (92903)</u>

In response to FNP's denial of this violation, the NRC reviewed the additional information and determined that this violation did not occur. NRC's letter, to Mr. D. Morey, SNC, from Mr. L. Plisco, Director of Reactor Projects, Region II, dated August 12, 1998, delineates the justification for withdrawing the violation.

E8.3 <u>(Closed) Inspector Follow-up Item (IFI) 50-348, 364/97-10-02: Updated Final Safety Analysis Report (UFSAR) Reverification Corrective Actions</u>

a. <u>Inspection Scope (92903)</u>

The inspector reviewed twelve 10 CFR 50.59 evaluation packages generated to resolve discrepancies identified by the UFSAR Verification program. The inspector also reviewed the applicable portions of the UFSAR and ASME Section III, Division 2, CC-3440, "Concrete Temperatures."

b. <u>Observations and Findings</u>

The inspector found that most packages were thorough and resolved the issue appropriately. However, package FVP-027 was identified which failed to identify an unresolved safety question (USQ). Package FVP-027 was reviewed and approved by the Plant Operations Review Committee (PORC) on December 18, 1997.

UFSAR Section 5.5.14.1.A, which described the reactor vessel support assemblies, stated "The supports are air-cooled to maintain the supporting concrete temperature at or below 130°F." The limit of 130°F met the temperature requirements for concrete specified in ASME Section III, Division 2, CC-3440, "Concrete Temperatures." Section CC-3440 stated that for normal operation or any other long-term period the concrete temperatures shall not exceed 150°F except for local areas, such as around a penetration, which are allowed to have increased temperatures not to exceed 200°F.

In 1977, the licensee identified that the required air flow through four of the six reactor vessel supports could not be maintained at 3000 standard cubic feet per minute (scfm). Based on measured airflows and the results of Calculation 9.1-11, dated July 14, 1977, the licensee determined that the concrete temperatures could not be maintained below the code limit of 150°F. The licensee concluded that the cooling air flow was adequate based on the 200°F local area limit allowed by the ASME code.

In 1997, The licensee's reverification process identified that the UFSAR was inconsistent with the actual concrete temperatures. A change to the UFSAR was proposed and approved by PORC to read: "The supports are air cooled to maintain the supporting concrete temperature at or below 190°F at a flow rate of 2000 scfm with an air temperature of 120°F to meet the acceptance criteria for localized concrete temperature of 200°F."

The 50.59 package for FVP-027 also relied on the ASME code 200°F local area exception to justify the UFSAR change. The inspector reviewed the code and discussed the purpose of the 200°F local temperature clause with NRR staff. The NRR staff determined it was not appropriate to use the 200°F temperature limit for the reactor vessel support concrete temperatures.

10 CFR 50.59 allows the licensee to make changes to the facility as described in the UFSAR, without prior NRC approval, unless the proposed change involves an USQ. A change is a USQ if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased. However, the licensee did not evaluate if the increased reactor vessel support concrete temperatures could increase the consequences of an accident previously evaluated in the UFSAR due to subjecting the concrete to elevated temperatures.

The inspector identified two additional UFSAR discrepancies during the review which were not identified by the licensee's UFSAR reverification.

- UFSAR Section 3.11.2.3 stated "Air temperature detectors will be located in the suction ducts of the reactor cavity cooling fans with indication and alarm in the control room." The inspector reviewed the plant documentation and interviewed licensee staff. No temperature detectors were located in the suction ducts of the reactor cavity cooling fans. The only temperature indicators for the reactor cavity that were indicated and alarmed in the control room were physically located in the reactor cavity and not in the suction ducts of the reactor cavity fans.

- UFSAR Section 6.2.3.2.4 stated "The reactor vessel support cooling system, consisting of two 100 percent capacity fans and ducting, is arranged to cool the reactor vessel supports by drawing air through the supports from the 155-ft elevation inside the containment, through the inspection opening above the reactor vessel nozzles." This was contrary to various plant drawings which indicates the cooling air flows up from the reactor cavity. The inspectors researched the issue and determined that it appeared cooling air flowed down from the 155 foot elevation, approximately 2000 to 4000 scfm, and up from the reactor cavity, 12000 to 16000 scfm.

These were identified to the licensee for resolution. In addition, these two discrepancies may also invalidate an assumption used to calculate the reactor vessel support temperatures. The calculations assumed that the temperature of the cooling air to the supports was 120°F. However, the inspector determined that the air temperature coming from the 155 foot elevation to the supports was at least 132°F. Also, since the initial temperature of the air in the reactor cavity could be greater than 95°F (based on the 95°F TS limit for Service Water temperature), the cooling air coming up from the cavity could be higher than 120°F due to absorbing heat while traveling past the thimble tubes and the reactor vessel insulation. The inspector provided these points to the licensee for resolution.

c. Conclusions

The 10 CFR 50.59 review of one UFSAR discrepancy package was inadequate and failed to identify an USQ. Also, the inspectors identified two additional UFSAR discrepancies that were overlooked by the licensee. This is identified as Escalated Enforcement Item (EEI) 50-348, 364/98-05-02, Failure to Identify Defacto 50.59 and USQ.

IV. Plant Support

R2 Status of RP&C Facilities and Equipment

R2.1 Radiologically Controlled Area (RCA) Tour (71750)

Overall cleanliness of the RCA remained good. Plant personnel observed working in the RCA demonstrated appropriate knowledge and application of radiological control practices. Health physics technicians provided positive control and support of work activities in the RCA. The Unit 1 containment was also toured during the forced outage. Conditions were noted as being good, with an appropriate level of HP oversight.

P1 Conduct of EP Activities

P1.1 Emergency Preparedness (EP) Drill (71750)

The inspectors observed and participated in an EP drill conducted on July 22, 1998. Activities in all emergency response facilities were observed. Emergency plan and procedure, and emergency action level declarations were verified to be correct. State and local county participation was also observed, and post-drill critique sessions were monitored. The inspectors reviewed drill related notifications and press releases.

The inspectors concluded that the EP drill was appropriately conducted, and was a good training exercise.

S1 Conduct of Security and Safeguards Activities

S1.1 Routine Observations of Plant Security Measures (71750)

The inspectors verified that portions of site security program plans were being properly implemented. Disabled vital area doors were properly manned and controlled. Security personnel activities observed during the inspection period were performed well. Site security systems were adequate to ensure physical protection of the plant.

S2 Status of Security Facilities and Equipment

S2.1 Security Facilities Walkdown (71750)

On July 30, 1998 the inspector walked down the site's security facilities including the alarm stations, the access portals and equipment, the protected and vital areas, perimeter intrusion detection systems, and other security related equipment. Security force members and management personnel were interviewed and activities were monitored.

The inspector noted the equipment and facilities were properly working, and security force members were alert and knowledgeable.

F1 Control of Fire Protection Activities

F1.1 Response to Fire Alarm in Document Control (71750)

The inspector observed the licensee's response to a smoke alarm and report of smoke in Document Control on August 20. The inspector concluded the licensee's response was in accordance with FNP-0-AOP-29.0, "Plant Fire," Rev. 18. The Shift Supervisor's command and control of the event response was good.

V. Management Meetings

X2 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 3, 1998. The licensee acknowledged the findings presented. No proprietary information was identified.

Partial List of Persons Contacted

Licensee

R. V. Badham, Safety Audit Engineering Review
C. L. Buck, Jr. Unit Superintendent
C. D. Collins, Operation Support Superintendent
R. M. Coleman, Maintenance Manager
G. P. Crone, Engineering Support Performance Supervisor
K. C. Dyar, Security Manager
T. H. Esteve, Planning and Control Superintendent
R. S. Fucich, Engineering Support Manager
S. Fulmer, Plant Training and Emergency Preparadness Manager
J. S. Gates, Administration Manager
D. E. Grissette, Assistant General Manager - Operations
J. G. Horn, Outage Planning Supervisor
J. R. Johnson, Operations Manager
D. H. Jones, SNC - Configuration Management Manager
R. A. Livingston, Chemistry Superisor

R. C. Lulling, Planning Supervisor
R. R. Martin, Maintenance Team Leader
M. W. Mitchell, HP Superintendent
R. L. Monk, Engineering Support Supervisor
C. D. Nesbitt, Assistant General Manager - Plant Support
J. E. Odom, Unit Superintendent
W. D. Oldfield, Nuclear Operations Training Supervisor
L. M. Stinson, Plant General Manager - FNP
R. J. Vanderbye, Emergency Prepardness Coordinator
G. S. Waymire, Technical Manager
R. L. Winkler, Engineering Group Supervisor, Plant Modification and Maintenance Support
B. R. Yance, Plant Modification and Maintenance Support Manager
M. Ajluni, Farley Nuclear Plant Licensing Manager, Southern Company Services (SCS)
J. Garlington, Steam Generator Replacement (SGR) Project Manager, SCS
S. Mayfield, Supervisor Nuclear Maintenance Support, SCS
D. McComb, SGR Engineering Manager, SCS
B. Moore, Manager Nuclear Maintenance Support, SCS
J. Thomas, SGR Quality Manager, SCS
R. Tyler, SGR Installation Manager, SCS

Other licensee employees contacted included construction craftsmen, engineers, technicians, operators, mechanics, and electricians.

List of Opened and Closed Items

Type	Item Number	Description and Reference
Opened		
EEI	50-348, 364/98-05-02	Failure to Identify Defacto 50.59 and USQ (Section E8.3).
Closed		
LER	50-348/97-06	Steam Generator Tube Degradation and Tube Status (Section 08.1).
LER	50-364/98-03	Steam Generator Tube Degradation and Tube Status (Section 08.1).
LER	50-348/97-05-01	TS Surveillances (Section 08.2).
LER	50-364/98-01	Manual Reactor Trip Due to Dropped Control Rod K-2 (Section 08.3).
LER	50-364/98-01-01	Manual Reactor Trip Due to Dropped Control Rod K-2 and F-10 (Section 08.3).
LER	50-348, 364/98-01	Inadequately Performed Surveillance Due To Improper Calculation of Average Disintegration Energy (E-Bar) (Section 08.4).

LER	50-348, 364/97-09	Lack of Missile Protection For Service Water Flow Switches (Section E8.1).
NCV	50-348, 364/98-05-01	Inadequate Missile Protection For Service Water Flow Switches (Section E8.1).
VIO	50-348, 364/97-14-05	Failure To Provide Tornado Missile Protection for TDAFW Pump Vent Stack (Section E8.2).
IFI	50-348, 364/97-10-02	Updated Final Safety Analysis Report (UFSAR) Reverification Corrective Actions (Section E8.3).

List of Inspection Procedures (IP) Used

IP 37550: Engineering

IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving, and Prevent Problems
IP 50001: Steam Generator Replacement Inspections
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Operation
IP 71750: Plant Support Activities
IP 90712: Inoffice Review of Written Reports
IP 92700: Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92903: Followup - Engineering
IP 92709: Licensee Strike Contingency Plans