U.S. NUCLEAR REGULATORY COMMISSION (NRC)

REGION II

Docket Nos:	50-348 and 50-364
License Nos:	NPF-2 and NPF-8
Report No:	50-348/97-05 and 50-364/97-05
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:	March 30 through May 10, 1997
Inspectors:	T. Ross, Senior Resident Inspector J. Bartley, Resident Inspector R. Caldwell, Resident Inspector (In training) G. Kuzo, Health Physics Inspector (Sections R1.1, R1.2, R2.1, R5.1, R8.1-8.5) W. Kleinsorge, Reactor Inspector (Sections M1.10 - M1.12) R. Chou, Reactor Inspector (Section E1.1) M. Ernstes, Operator Licensing Examiner (Section O.5)
Approved by:	P. Skinner, Chief, Projects Branch 2 Division of Reactor Projects

Enclosure 2

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EXECUTIVE SUMMARY

Farley Nuclear Power Plant, Units 1 and 2
NRC Inspection Report 50-348/97-05, 50-364/97-05

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

Operations

- Operator attentiveness to MCB annunciator alarms and response to changing plant conditions were prompt. Interviews with members of the operating crew revealed that they were consistently aware of plant conditions and ongoing activities (Section O1.1).
- Unit 1 refueling activities were well-controlled (Section O1.2).
- Housekeeping and physical conditions were adequate, although certain areas remained poor. Licensee efforts to improve targeted areas was evident. Improvement in overall plant appearances and material conditions has increased (Section O2.1).
- Safety system walkdowns and tours verified accessible portions of selected systems were well maintained and operational (Sections O2.1 and O2.2).
- Tag orders were properly executed, with one identified exception (Section O2.3).
- A violation was identified for failure to notify the NRC of a change in a licensed Senior Reactor Operator's medical status (Section O5.1).
- Licensee efforts to identify, resolve, and prevent problems remained effective, with one identified exception (Section O7.1).

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 (Sections M1.1 through M1.7).
- Over the years, numerous foreign objects and materials have been allowed to enter the emergency core cooling system containment sumps. Licensee efforts to clean, inspect, and repair the sumps have been very thorough to date (Section M1.7).

- A violation was issued for failing to include several specific Technical Specifications (TS) surveillance requirements in the Surveillance Test Program. Examples of other missed TS surveillance requirements have been identified in recent reports. Load rejection testing of emergency diesel generators (EDGs) was inconsistent with the licensing basis (Section M1.8).

- The inspectors' issues regarding compliance and conformance with the licensing basis of TS 4.8.1.1.2.e are identified as unresolved item (URI) 50-348, 364/97-05-04, EDG 50% Load Reject Surveillance Testing, pending the NRC's response to the licensee's TS interpretation request and proposed TS amendment (Section M1.8).

- Inservice inspection activities observed/reviewed were conducted in accordance with procedures, licensee commitments and regulatory requirements (Section M1.10).

- A violation was identified associated with the control of the special process of welding (Section M1.11).

- A weakness was identified associated with the licensee's control of contracted welders (Section M1.11).

- Maintenance was subjected to independent audits, with appropriate action generally taken for identified weaknesses (Section M1.12).

- The issue of control of painting inside the penetration room boundary (PRB) being limited to less than 1000 ft^2 in any 24 hour period was not documented by an existing analysis. This is identified as URI 50-364/97-05-06, Painting Effects On PRF Operability, pending review of the licensee's analysis (Section M8.1).

Engineering

- Calculations analyzing the SG loop C narrow support gap on Unit 1 were acceptable. Licensee promptly evaluated the potential for past damage and restored the intended conditions by modifications (Section E1.1).

- Penetration room filtration (PRF) system licensing bases operability is identified as URI 50-348, 364/97-05-07, Licensing Basis for PRF System During Post-LOCA Recirculation, pending additional review by the NRC (Section E1.2).

Plant Support

- Radiological controls were good for routine operations and U1RF14 outage activities. All personal exposures were within 10 CFR Part 20 limits. Implementation of established ALARA program activities was verified (Section R1.1).

- Service air compressor system supplied Grade D respirable air in accordance with 10 CFR 20, Appendix A requirements (Section R1.2).
- Training and medical certifications for personnel using respiratory protective equipment were conducted in accordance with the licensee procedures and met the applicable requirements of 10 CFR Part 19 and 10 CFR Part 20 (Section R5).
- Health Physics control over the radiologically controlled area, and the work activities conducted within it, was good. Some contaminated areas were cramped and physically restricted removal of anti-contamination clothing (Section R2.1).
- Security activities continued to be performed in a conscientious and capable manner, assuring the physical protection of protected and vital areas. Problems with improper display of security badges by workers in containment were resolved promptly (Section S1.1).
- Significant weakness identified in the implementation of Open Flame Permits (Section F1.1).
- Life safety exits from the Turbine Building were inadequate (F2.3)

Report Details

Summary of Plant Status

Unit 1 was shutdown for its 14th refueling outage (U1RF14) during the entire inspection report period. U1RF14 was rescheduled to be completed in 65 days instead of the original 55 days due to increased scope of steam generator (SG) tube inspection and repair work.

Unit 2 operated continuously at 100% power for the entire inspection period, except for a brief power reduction to 85% power during the weekend of May 3, 1997, for main turbine generator (MTG) governor valve testing.

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 Specifications (TS) Limiting Condition of Operation (LCO) tracking sheets, walked down the Main Control Boards (MCB), and interviewed members of the operating shift crews to verify operational safety and compliance with TSs. The inspectors 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 safety concerns.

b. Observations, Findings and Conclusions

Overall control and awareness of plant conditions during the inspection period remained adequate. Inspectors observed that the Unit 2 MCBs and emergency power board were nearly "blackboard" most of the inspection period, with only a couple of persistent annunciators in alarm that were recognized deficiencies. Efforts to maintain MCB deficiencies at low levels continued. MCB deficiencies on Unit 2 increased to 15 or more and have remained there. Almost all of them involved nonsafety-related instrumentation or equipment. Operator attentiveness to MCB annunciator alarms and response to changing plant conditions were prompt. Interviews with members of the operating crew revealed that they were consistently aware of plant conditions and ongoing activities. Pre-shift briefs of the operating crews by the shift supervisors (SS) were generally concise and informative. Operator logs generally were of sufficient detail and scope, with one notable exception. Surveillance test procedure (STP) FNP-1-STP-40.1, Revision 27, "B1F and B1H Sequencer Load Shedding

Circuit Test," was performed on April 12, 1997, but was not mentioned in the Unit 1 reactor operator (RO) logs. This omission was discussed with Operations management.

O1.2 Unit 1 Refueling (IP 60710)

A resident inspector observed refueling activities from the MCR, Spent Fuel Pool (SFP), and MCR balance of plant area, on 18 April, 1997. The Westinghouse Refueling Manual FP-ALA-R14, Revision 0, J.M. FARLEY Unit 1, Cycle XIV - XV Refueling, conducted under WA# W00475930, was reviewed. Refueling activities observed by the resident inspectors were performed in a well controlled and methodical manner in accordance with FNP-1-UOP-4.1, "Controlling Procedure for Refueling and the Westinghouse Refueling Manual," Revision 9. Communications between the various stations were clear and concise. The subcritical multiplication plot (1/M plot) was accurate and timely. Personnel were very familiar with the procedure and knowledgeable of the process and systems. No significant incidents occurred during fuel handling and all observed fuel assemblies were landed in their appropriate locations. The inspector concluded that fuel handling was accomplished in a professional and competent manner.

O2 Operational Status of Facilities and Equipment

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

General tours of safety-related areas were performed by the inspectors 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 in the following areas:

- Unit 1 containment
- Unit 1 and 2 SFP, SFP heat exchangers (HXs), and SFP cooling pump rooms
- Unit 1 and 2 main steam valve rooms
- Unit 1 and 2 piping penetration rooms (PPR) on 100 foot elevation
- Unit 1 and 2 PPRs on 121 foot elevation
- Service water intake structure (SWIS)
- Unit 1 component cooling water pump and heat exchanger (HX) rooms
- Unit 1 and 2 vital 125 volt direct current (VDC) switchgear and battery rooms
- Unit 1 and 2 new fuel storage areas
- Unit 1 and 2 service and instrument air compressors, dryers and receivers
- Emergency diesel generator (EDG) building
- Unit 1 and 2 containment spray (CS) pump rooms
- Unit 1 and 2 residual heat removal (RHR) HX rooms
- Unit 1 and 2 RHR pump rooms
- Unit 1 and 2 charging pump rooms and hallway

- Turbine building
- Unit 1 and 2 penetration room filtration (PRF) system rooms
- Unit 2 hot shutdown panel (HSDP) rooms
- Unit 1 filter gallery on 139-foot elevation
- Unit 1 vital 4160 volt alternating current (VAC) switchgear rooms, trains A and B, including vital 600 VAC load centers

General material conditions and housekeeping for Unit 2 were adequate. Areas were generally clear of trash and debris. Attempts to control the impact of Unit 1 outage activities were obvious, as were additional efforts at housekeeping in the outage unit. Considerable effort to improve physical appearances of plant areas and equipment was in progress, primarily in the form of extensive painting in the EDG and auxiliary buildings. These efforts have dramatically improved the appearances of rooms, structures and equipment in targeted areas. 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. Maintaining all critical areas of the auxiliary building well lamped remained an ongoing challenge. None of these problems represented operability concerns.

O2.2 Biweekly Inspections of Safety Systems (IP 71707)

Inspectors used IP 71707 to verify the operability of the following selected safety systems and/or equipment:

- Unit 2 Hot Shutdown Panels
- Unit 1 Accumulators
- 1B Emergency Diesel Generator
- Unit 1 Containment Spray, Trains A and B

Accessible portions of the systems listed above were verified to be properly aligned and appeared to be well maintained and in good operating condition. The inspectors did not identify any significant issues that adversely affected system operability. In addition, accumulator pipe supports and components were walked down using isometric system drawings.

O2.3 Tag Orders (IP 71707)

During the course of routine inspections, portions of the following tag orders (TO) and associated equipment clearance tags were examined by the inspectors:

- TO# 97-0757-1; RHR
- TO# 97-1059-1; 1B PRF EQ Upgrade
- TO# 97-1058-1; 1A PRF EQ Upgrade
- TO# 97-442-1; Service Water System (SWS) pumps 1C, 1D, and 1E
- TO# 97-178-1; CS System and Chemical Addition Tank
- TO# 97-1165-1; SW to Containment Coolers 1A and 1B

All tags and TOs examined by the inspectors were properly executed and implemented. However, one tag order was identified by the inspectors that used an air-operated valve, which fails to the open position, as part of the clearance boundary.

While walking down TO 97-1059 on May 8, 1997, the inspector noted that Q1V48HV3538B, SFP to 1B PRF Supply Damper, was tagged shut per the tag order. This valve is air-operated and thus depended on air pressure to maintain itself in the tagged closed position. Furthermore, because of the type of work being done on the 1B PRF filter housing, this valve was required to remain closed to ensure the 1A PRF system was not rendered inoperable by short-circuiting the suction flow path from the spent fuel pool. The inspector interviewed the TO author and determined that he was not aware that HV3538B was a fail open valve nor was he aware of the potential for compromising the operability of the other PRF train. During discussions with additional operations personnel, the inspector determined that other operations staff were also not aware of the potential for short-circuiting the SFP supply to the 1A PRF system if HV3538B had failed open. When informed of the inspector's concern regarding operability of the 1A PRF system, the SS promptly revised the TO to jack shut HV3538B and initiated OR 1-97-191. The inspector determined, through interviews with licensee personnel that loads were not moved in the spent fuel pool area while either 1A PRF or 1B PRF were tagged out for the environmental qualification upgrade. Therefore, no TS LCO action statements were required to be entered while the PRF was potentially inoperable.

O2.4 TS LCO Tracking (IP 71707)

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

O5 Operator Training and Qualifications

O5.1 Conditions of Operator Licenses

a. Scope (IP 71001)

The inspectors reviewed 10 CFR 55 and NRC Information Notice 94-14, "Failure to implement requirements for biennial medical examinations and notification to the NRC of changes in licensed operator medical conditions," which was issued to remind licensees of the requirements to notify the NRC of changes in a licensed operators physical or mental condition. 10 CFR 55.21, Medical examination, requires an NRC-licensed operator to be examined by a physician every 2 years to determine if the individual meets the requirements of 10 CFR 55.33(a)(1). If, during the term of the license, the operator develops a permanent physical or mental condition that causes the operator to fail to meet the

requirements of 10 CFR 55.21, the facility licensee shall, in accordance with 10 CFR 55.25, notify the Commission within 30 days of learning of the diagnosis.

b. Observations and Findings

On April 9, 1997, an NRC inspector reviewed applications for renewal of Senior Reactor Operator licenses. The NRC Form 396, "Certification of Medical Examination by Facility Licensee," of one of the applicants indicated that he needed corrective lenses to meet the requirements of 10 CFR 55.33(a)(1). However, the individual's current Senior Reactor Operator license contained no restrictions. Based upon a telephone conversation with Farley Training Department personnel, it was determined that the individual had been diagnosed as needing glasses during a biennial physical conducted on March 21, 1995. However, the facility licensee failed to notify the NRC of the change. The licensee conducted a review of all other licensed operators' medical conditions to determine if any other changes in medical conditions had failed to be reported. None were found.

c. Conclusion

The licensee's failure to notify the NRC within 30 days of a permanent change in a licensed operator's medical status is identified as violation (VIO) 50-348, 364/97-05-01, Failure to Notify NRC of Change of Licensed Operator Medical Status.

O7 Quality Assurance in Operations

O7.1 Effectiveness of Licensee Control in Identifying, Resolving, and Preventing Problems (IP 71707 and 40500)

The inspectors briefly reviewed all newly initiated ORs and completed ORs approved during the inspection period to ensure that plant incidents which affect or could potentially affect safety were properly documented and processed in accordance with Administrative Procedure (AP) FNP-0-AP-30, "Preparation and Processing of Incident Reports," Revision 22. Selected ORs that had been completed were reviewed in detail.

The inspectors concluded that the licensee's program for identifying and resolving problems remained effective and was being accomplished in accordance with FNP-0-AP-30. Plant personnel and management exhibited an appropriate threshold for identifying problems, initiating ORs, and assigning formal root cause determinations. Each new OR received prompt attention and was discussed in the morning status/plan of the day meeting.

Inspectors reviewed the following ORs for accuracy, completeness and reportability, and adequacy of corrective actions:

- OR #1-97-132: Non-temperature compensated Heise gauges
- OR #1-97-191: PRF TO deficiency
- OR #1-97-198: DG 1B Rollup door stuck open

These ORs were properly processed to completion. In general, licensee corrective actions for resolving problems continued to remain effective. However, OR 97-132 was initiated to address an instance where approved corrective actions were not fully implemented. On April 9, 1997, while touring the Unit 1 Auxiliary Building, the inspector found four non-temperature compensated 0 - 100 psig Heise test gauges which were not labeled with a restricted use tag. The gauges were staged for use in local leak rate testing. The use of non-temperature compensated Heise gauges without restricted use tags was contrary to the corrective actions identified in Corrective Action Request (CAR) 2173 for VIO 50-348, 364/95-18-03. Licensee staff immediately impounded the four gauges when informed by the inspector and initiated OR 1-97-132. A re-audit identified four additional non-temperature compensated Heise test gauges in the Calibration Lab. The licensee investigation determined that these eight gauges, which were used by the Systems Performance Group for local leak rate testing (LLRT), were stored under a green tarp in a mechanical room. They were overlooked when the actions for CAR 2173 were performed. As further corrective action for this event, the licensee: 1) performed two independent audits which did not identify any more test equipment requiring restricted use tags due to lack of temperature compensation, 2) modified calibration equipment checkout procedures to check for temperature compensation, and 3) evaluated all work activities which used any of these eight gauges. The inspectors independently reviewed work activities which used these gauges and determined that they were not invalidated by the use of a non-temperature compensated gauge. Failure to fully implement all of the corrective actions identified by CAR 2173 and NOV response letter dated December 19, 1995 constitutes a minor violation consistent with the guidelines of Section IV of the NRC Enforcement Policy. This non-cited violation (NCV) is identified as NCV 50-348, 364/97-05-09, Failure to Fully Implement Corrective Actions.

O8 Miscellaneous Operations Issues (92901)

O8.1 <u>(Closed) LER 50-348/97-002: Safety-Related 4160 Volt AC Breakers Not Seismically Qualified</u>

<u>(Closed) LEP 50-348/97-004: Safety-Related 600 Volt AC Breakers Position Sensitive Seismic Qualification</u>

Inspectors verified that 4160 and 600 Volt AC safety-related breakers that can not be maintained in their seismically qualified positions were removed from their respective cubicles, placed an appropriate distance

away, and properly secured. A few minor discrepancies (e.g., lack of blocking bolts on 4160 VAC breaker assembly wheels) were initially identified by the inspectors and promptly corrected. No repeat problems were found. The inspectors also reviewed applicable operating procedure FNP-0-SOP-36.6, "Circuit Breaker Racking Procedure," Revision 12 and Temporary Change Notice (TCN) 12A, and interviewed responsible personnel regarding proper handling and positioning of these breakers. One minor procedure deficiency was identified and discussed with Operations management regarding inadequate instructions for removing 600 volt breakers and placing them a prescribed distance away. Although different than the corrective actions described in LER 97-004, Operations has decided it would rather place 600 volt load center breakers in spare cubicles rather than on the floor, whenever possible. By the end of the report period, applicable procedures were being revised to allow this option. These LERs are closed.

O8.2 <u>(Closed) LER 50-364/96-001: Reduction/Resumption of 2B Diesel Generator Speed Caused By Inadequate Procedural Guidance</u>

This LER was considered a minor issue and closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 <u>General Comments</u>

a. <u>Inspection Scope (IP 61726 and 62707)</u>

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):

- WO# M96004128 — ICCMS Probe Changeout
- WA# 00467851 — Q1P16MOV3019B MOVATS testing
- WO# 00448943 — Sequencer Panel B2H Sequence Start Undervoltage Relay 0A-C
- WO# 00448910 — Sequencer Panel B1H Sequence Start Undervoltage Relay 0B-C
- WO# 00448913 — Sequencer Panel B1J Sequence Start Undervoltage Relay 0B-C
- WO# S00080418 — 1B PRF EQ Upgrade
- FNP-1-STP-80.11 — DG 1B 1200 KW Load Rejection Test, Revision 7

• FNP-2-STP-16.2	Containment Spray Pump 2B Inservice Test, Revision 23
• FNP-0-MP-14.19	Emergency Diesel Generator 1-2A, 1B, and 2B Removal and Inspection of Air Start Check Valves, Revision 2
• FNP-0-IMP-226.7B	Diesel Generator Single Circuit Emergency Start Test, Revision 7
• FNP-0-SOP-38.0	Diesel Generators, Revision 56
• FP-ALA-R14	J.M. FARLEY Unit 1, Cycle XIV - XV Refueling, Revision 0, FCR 08, Westinghouse Refueling Manual
• WO #S00080409	1A PRF EQ Upgrade, DCP No. B97-1-9157
• WO #S00080408	1A PRF EQ Upgrade, DCP No. B97-1-9157
• WO #S00080411	Rewire 1A PRF heater coil, DCP No. B97-1-9157
• WO #S00080410	Rework 1A PRF fan motor Q1E15M001A(2A), DCP No. B97-1-9157
• FNP-1-STP-228.5	NIS Power Range Channel N41 Calibration N1C55NE0041, Revision 38
• FNP-1-STP-40.0	Safety Injection with Loss of Off-Site Power Test, Revision 31
• FNP-1-STP-40.0 (Appendix S)	Retest of Q1P16MOV3134, SW from RCP MTR CLRS under WO #97003589
• FNP-1-STP-256.15	Loss Of Offsite Power Response Time Test, Train A, Revision 14 (TCN 14B) under WO #00467991
• FNP-1-STP-16.1	Containment Spray Pump 1A Inservice Test, Revision 29 under WO #0079666
• FNP-0-MP-14.1	Emergency Diesel Generator 1-2A, 1B & 2B Refuel (18 Month) Inspections, Revision 23
• FNP-0-MP-84.4	Emergency Diesel Generator Vibration Measurements, Revision 4
• FNP-0-SOP-42.0	Diesel Generator Fuel Oil Storage & Transfer System, Revision 2C
• FNP-2-STP-24.1	2A, 2B, and 2C Service Water Pump Quarterly Inservice Test, Revision 23
• WO #S0068708, 11 and 12	SW Supply and Return Valve Replacements for Containment Coolers 1A and 1D
• FNP-1-STP-80.14	Diesel Generator A Train Loss of Offsite Power Test
• FNP-1-STP-80.15	Diesel Generator B Train Loss of Offsite Power Test
• WA #W00477033	Preventive Maintenance (PM) Lubrication of 2C Charging Pump

b. Observations, Findings and Conclusions

All of the maintenance work and surveillance testing observed by the inspectors was performed in accordance with work instructions, procedures, and applicable clearance controls. No adverse findings were identified. Safety-related maintenance and surveillance testing

evolutions were well planned and executed. 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 MCR. Overall, operators and technicians appeared knowledgeable, experienced, and well trained for the tasks they performed.

M1.2 1B D/G Outage Maintenance

The inspector observed portions of the 18 Month 1B EDG maintenance and testing work from March 30 - April 4, 1997. Specific evolutions observed include portions of the following:

- Removal and inspection of air start check valves.
- Replacement of Lube Oil HX and Jacket Water HX tube bundles due to inlet tube sheet erosion concerns.
- Diesel Generator circuit emergency start inspection done under WO #470331.
- Various other refuel (18 Month) inspections.
- Emergency Diesel Generator vibration measurements.
- Inspection of the Diesel Generator fuel oil storage and transfer system.

All work was completed per procedure in a professional manner. Post maintenance testing was completed satisfactorily.

M1.3 Manual Safety Injection (SI) Hand Switch Test

The inspectors observed the test in the MCR and at the reactor trip breakers. The test was performed in accordance with FNP-1/2-STP-38.3, "Verification of Manual SI Actuation Input," Revision 1. All components responded as required. Refer to section M8.2 for more details.

M1.4 Unit 1 Gammametrics Probe Replacement

The inspectors observed maintenance activities involving the pre-job planning, removal, and installation of the Gamma-Metrics Detector, Q1C55NF0048. This work was performed under WA #00464676. Pre-job planning was thorough. Licensee personnel were knowledgeable and sensitive to ALARA planning. Maintenance and HP interfaced well to minimize radiation exposures. Work was conducted per the WA and the staff responded well to unexpected situations such as the detector being much more activated than expected.

M1.5 Unit 1 Containment Cooler Coil Replacement

An inspector observed portions of the work to remove all the original containment cooler coils and replace them with new coils according to design change package (DCP) 96-1-9054. All work witnessed by the inspector was accomplished in accordance with authorized work orders. Good cleanliness control was exhibited to prevent introducing foreign material into the cooler piping. During modification work, the licensee's activity task manager identified two problems that warranted being documented as ORs: 1) Original cooler coils were missing mounting bolts that attached the coils to the cooler frame (OR# 97-94) and 2) Baffle gaskets were missing from some of the replacement coils (OR# 97-151). The inspector discussed each of these problems in detail with the activity manager. After the containment cooler coils were replaced, and the system returned to service, the inspector conducted a walkdown of all four containment coolers. The inspector also reviewed the post-modification testing (PMT) performed in accordance with FNP-1-STP-17.0, "Containment Cooling System Train A(B) Operability Test." Although both trains of containment coolers met the acceptance criteria of STP-17.0, the inspector discussed testing the coolers individually in order to verify sufficient SW flow through each one. After further review, the licensee expanded the PMT to measure flow through each cooler per WO #S97004055. The inspector reviewed the results which verified adequate flow.

M1.6 Unit 1 CS Addition Tank Deletion and Trisodium Phosphate Basket Installation

On May 8, a resident inspector observed the visual examination (i.e., VT-2) performed on welds associated with the Unit 1 "A" train Spray Additive Tank removal design change, under WO# 0079666. The 1A Containment Spray pump was started per FNP-1-STP-16.1, "Containment Spray Pump 1A Inservice Test," Revision 29, and the weld checked at operating pressure. A quality control (QC) inspector conducted a detailed visual check of the weld area and blank flange. The QC inspector and resident inspector observed no leakage.

A resident inspector conducted a walkdown of the new trisodium phosphate (TSP) baskets built on the containment basement floor in accordance with DCP 95-1-8931. The inspector also observed licensee efforts on May 13 to load 70 fifty pound bags of TSP into each of the three baskets. Upon the completion of this work the inspector noticed and discussed the following concerns with the responsible design change engineer: 1) All three TSP baskets were slowly leaking at the bottom corners, 2) Actual level of TSP in the baskets did not rise to the minimum TS level marked on the baskets, and yet the quantity loaded should have been sufficient, 3) Accuracy of TSP bag weight, and 4) Maintaining inventory control of material lost during basket leak/repair. Each of these observations was adequately addressed by the licensee. Subsequent walkdowns of the

finished and filled baskets by the inspector did not identify any additional concerns.

M1.7 Emergency Core Cooling System Containment Sump Inspections and Foreign Material Retrieval

On April 24, 1997 an SAER auditor and PMD engineer discovered that the Unit 1 emergency core cooling system (ECCS) sump screens were not built per design. Two of the four ECCS sumps in containment were missing a structural, flat bar of steel. OR #97-172 was initiated. Thorough walkdowns of the Unit 1 ECCS sumps conducted by an inspector and licensee personnel also identified numerous gaps in the sump screens created by poor fitup. Many of the gaps were on the order of 1" x 1," which are considerably larger than the 1/8" x 1/8" screen mesh openings. On May 2, the licensee made a containment entry into Unit 2 to examine its ECCS sumps. Although gaps of comparable size were also discovered in the Unit 2 screens, they were much fewer in number. This inspection also included the interior of the Unit 2 sumps and identified two pieces of foreign material that were removed. OR #97-183 was initiated for Unit 2. The Unit 2 ECCS sump screen gaps were promptly repaired, returning the ECCS sumps back to original design. An operability determination (OD) 97-10 was developed by the licensee and reviewed by the inspector. It determined that operability of ECCS equipment had not been adversely affected. The inspector discussed with the licensee the potential that foreign material may have also fallen down inside the ECCS sump suction piping for the CS and RHR systems.

Beginning May 8, the inspector observed maintenance personnel conduct internal visual inspections of the Unit 1 ECCS sump, both the interior grating and suction piping. A boroscope was used to inspect inside the suction piping. Numerous articles of foreign debris were discovered inside the interior grating (e.g., many different lengths of duct tape, metal washers, marking pen, weld rod segment, etc.) and sump suction piping (e.g., many different lengths of duct tape, short piece of string, small strip of plastic, etc.). The boroscope used by the maintenance personnel could only inspect about six feet of the horizontal run of piping. By the end of the inspection period, the licensee was still cleaning out the foreign debris from the Unit 1 sumps and enlisting the aid of a contractor who could exam the entire run of suction piping. Operations was also pursuing an OD to address the operability impact of the discovered material in Unit 1. Plans for conducting internal visual examination of Unit 2 interior grating and sump suction piping was still being discussed. In order to track the licensee's ongoing corrective actions and operability evaluations this issue is identified as inspector followup item (IFI) 50-348, 364/97-05-02, Foreign Material In Containment ECCS Sumps.

M1.8 Failure to Meet Multiple TS Surveillance Requirements

During the current Unit 1 outage (U1RF14), the licensee identified the following TS Surveillance Requirements that were not being adequately implemented and/or missed entirely:

a) TS Table 4.3-1, Reactor Trip Instrumentation Surveillance Requirements, Functional Unit 6 requires a quarterly channel functional test and shiftly channel checks of the nuclear instrumentation system (NIS) source range (SR) channels while in Modes 2, 3, 4, and 5. On March 19, 1997, during a review of its Improved TS submittal, the licensee determined that when Unit 1 was shutdown on March 15 for U1RF14, the unit had entered Mode 2, and then several minutes later Mode 3, without accomplishing the surveillance requirements for NIS SR. The TS surveillance requirements were not met until March 18 while the unit was in Mode 5 after NIS SR N-32 was functionally tested. OR# 97-139 was initiated and LER 97-05 was written to document, resolve and report the incident. Since the root cause was considered to be inadequate test procedures, the licensee also concluded that the failure to meet these TS surveillance requirements has occurred numerous times in the past during previous Unit 1 and 2 shutdowns.

b) TS Table 4.3-1, Functional Unit 2.B, for NIS power range (PR), neutron flux-low, requires quarterly channel calibration while in Mode 2. On April 8, 1997, while conducting a review specifically looking for TS mode change difficulties, the licensee determined that Unit 1 entered Mode 2 on March 15 without accomplishing the surveillance requirements of TS Table 4.3-1. This incident was also reported in LER 97-05. Unit 1 was only in Mode 2 for about seven minutes until it entered Mode 3. Due to inadequate test procedures and TS inconsistencies, the licensee concluded this TS surveillance requirement had been missed during other Unit 1 and 2 shutdowns.

c) On April 23, 1997, while conducting a special, comprehensive review of TS mode change requirements, the licensee determined that the 18 month TS surveillance requirement 4.8.1.1.2.c.8 had not been performed on Unit 2 for EDG 1-2A or 1C. OR# 97-162 was initiated and LER 97-03 (Unit 2 only) was written to document, resolve and report the incident. TS 4.8.1.1.2.c.8 requires verifying a simulated SI signal will override the test mode of an operating EDG connected to its bus returning it to standby operation. This surveillance is a TS requirement for both units. However, FNP-0-STP-80.8, "Diesel Generator 1-2A 1000 KW Load Rejection Test," and FNP-0-STP-80.9, "Diesel Generator 1C 1000 KW Load Rejection Test," which perform the required surveillance test, were only done on Unit 1. Credit was then given for meeting the TS surveillance requirement for both units. On April 24, the 1-2A EDG was successfully tested while connected to its vital 4160

VAC bus on Unit 2 (i.e., Bus 2F). The 1C EDG was successfully tested on Unit 2 the following day.

Although these were identified by the licensee, these are being cited as a violation due to their repetitive nature and inadequate licensee corrective action. These examples of failure to follow TS surveillance requirements constitute a violation identified as VIO 50-348, 364/97-05-03, Failure To Follow Multiple TS Surveillance Requirements.

On April 25, 1997, the licensee also examined its compliance with TS surveillance requirement 4.8.1.1.2.e which requires conducting a load rejection test of 1200 - 2400 KW every five years without tripping the EDG, and verifying that all fuses and breakers on the energized emergency bus(es) are not tripped. Similar to example c) above, the licensee discovered that the applicable FNP-0-STP-80.11, "Diesel Generator 1-2A 1200 KW Load Rejection Test," Revision 8, and FNP-0-STP-80.12, "Diesel Generator 1C 1200 KW Load Rejection Test," Revision 6, were not being performed on both units. Instead, STP-80.11 was routinely conducted only on Unit 1 and STP-80.12 was being conducted on Unit 2. But, after additional evaluation, the licensee concluded that TS 4.8.1.1.2.e could be satisfied for the shared A train EDGs by only testing them on one unit and taking credit for both. A resident inspector reviewed the licensee's evaluation and disagreed with its conclusion. The inspector read the statement in TS 4.8.1.1.2.e, for both units, that says "Verify that all fuses and breakers on the energized emergency bus(es) are not tripped" seems to imply this surveillance test is not just an EDG test. Rather, each unit's emergency bus(es) are also required to be tested during an EDG load rejection to ensure their respective breakers and fuses do not trip. The inspector conferred with NRC technical staff regarding this difference in opinion, and then notified the licensee that their position appeared inconsistent with TS requirements. Following discussions with the inspector, SNC concluded that a differing opinion continued to exist. To reconcile this difference SNC formerly requested a TS interpretation from the NRC by letter dated May 22, 1997. In addition, SNC also conducted a successful 1220 KW load rejection test of the 1C EDG tied to its Unit 1 emergency bus; and depending on Unit 2 availability, the licensee was considering testing the 1-2A EDG. While awaiting NRC response to the May 22 letter, the licensee also has submitted a TS amendment request dated May 28, 1997, that clarifies TS 4.8.1.1.2.e.

In addition to the compliance implications of TS 4.8.1.1.2.e discussed above, the inspector also questioned the licensee on the adequacy of the surveillance testing being conducted from a technical and licensing basis. Both STP-80.11 and 80.12 only require their respective EDG to undergo a 1200 KW load rejection test, which is the minimum allowed by TS. The TS load rating of the 1C EDG is 2850 KW and the 1-2A EDG is 4075 KW. According to the NRC's safety evaluation report (SER) that approved the TS 4.8.1.1.2.e surveillance requirement, "the licensee has

devised a test that, by manually tripping two circuit breakers (leaving the diesel generator output breaker closed), a load approaching 50% of rating is rejected ..." The SER also stated, "the confidence gained from...testing of loss of half the rated load every five years is sufficient to provide reasonable assurance on a continuing basis that the diesel generator will not be lost due to a load rejection..." A 50% load rejection for the 1C and 1-2A EDG would be 1400 KW and about 2030 KW, respectively. TS 4.8.1.1.2.e establishes a range that specifically allows for testing each EDG at their own 50% load rating. However, contrary to statements in the SER, the licensee's STPs only test the 1-2A and 1C EDGs at 29% and 42% of their load rating, respectively. STP-80.11 and 80.12 only trip one circuit breaker to initiate the load rejection vice two described in the SER. The inspector discussed these concerns at length with SNC management. The licensee will evaluate the adequacy of surveillance testing methodology in light of the current licensing basis and technical arguments.

These concerns are identified as unresolved item (URI) 50-348, 364/97-05-04, EDG 50% Load Reject Surveillance Testing, pending NRC's response to the licensee's TS interpretation request and proposed TS amendment.

M1.9 SI/LOSP Integrated Test

On April 25, 1997, a resident inspector reviewed the results of FNP-1-STP-40.0, "Safety Injection with Loss of Off-Site Power Test," Revision 31. The test appeared to be challenging and complex. Test completion was considered satisfactory with several relatively minor exceptions, which were noted and rescheduled. Test signoffs were current and the pre-job checklist was complete. Test data packages appeared to be complete and results met the defined acceptance criteria. On May 5, a resident inspector observed FNP-1-STP-40.0, Appendix S, "Retest of Q1P16MOV3134, SW from RCP MTR CLRS," conducted under WO 97003589. The pre-job brief conducted by the Shift Supervisor was clear, concise and identified potential areas of concern. The retest was conducted satisfactorily after a procedural inaccuracy was reconciled. Overall, the test and retests appeared to adequately verify SI operation with a loss of offsite power.

M1.10 Inservice Inspection

a. Inspection Scope (IP 73753)

To evaluate the licensee's Inservice Inspection (ISI) program and the program's implementation, the inspectors reviewed procedures, observed work in progress and reviewed selected records. Observations were compared with applicable procedures, the Final Safety Analysis Report (FSAR) and ASME B&PV Code Sections V and XI, 1983 Edition, Summer 1983 Addenda (83S83).

Specific areas examined included the following: observation of Liquid Penetrant (PT) examination of Item Nos. ALA2-4517-37 and ALA2-4516-1; manual Ultrasonic (UT) examination of Item No. ALA2-4516-1; data acquisition and analysis activities associated with Eddy Current (ET) examinations of Steam Generator (S/G) tubing; data acquisition and analysis activities associated automated UT examinations of reactor vessel welds; review of video tape of the remote Visual (VT) examination of the reactor vessel internals; direct VT examination of support ALA2-4516-SI-R206; Review of selected completed examination reports; and review of the Repair and Replacement Program.

The inspectors performed an independent evaluation of indications to confirm the licensee's ISI examiners' evaluations.

The inspectors reviewed records for the Nondestructive Examination (NDE) personnel and equipment utilized to perform ISI examinations. The records included: NDE equipment calibration and materials certification; and records attesting to NDE examiner qualification, certification and visual acuity.

The inspectors scrutinized FNP Occurrence Report OR No. 1-97-115.

b. Observations and Findings

ISI examinations observed/reviewed were conducted in accordance with properly approved procedures, by qualified and properly certified examiners using properly certified/calibrated equipment and materials.

The licensee had implemented the Containment Inspection Rule repair and replacement (R/R) program by revisions to FNP-1-M-043, "Second Ten-Year Inservice Inspection Program For Class 1, 2, and 3 Components" and FNP-0-GMP-0.2, "Repair and Replacement Instructions for ASME Class 1, 2, 3, and MC Components," and the issuance of FNP-0-GNP-0.4, "Repair and Replacement Instructions for ASME Class CC," FNP-0-100-.24, Visual Examination VT-1 For IWE Components," FNP-0-100-.25, "Visual Examination VT-3 For IWE Components," FNP-0-100-.26, "Visual Examination VT-1C," FNP-0-100-.27, "Visual Examination VT-3C."

FNP-1-M-043, paragraph 1.9 states, in part, "Code Case N-416-1 (RR-47) may be used in place of hydrotest in some situations. The NRC SER requires a surface examination on the root pass layer of Class 3 butt and socket welds on pressure retaining boundaries." The NRC SER had not been identified and the root inspection option had not been addressed in lower tier documents. The licensee indicated that they would take appropriate action.

The issue documented in OR No. 1-97-115, was properly identified, evaluated and closed out.

c. Conclusion

ISI activities observed/reviewed were conducted in accordance with procedures, licensee commitments and regulatory requirements.

M1.11 Welding

a. Inspection Scope (IP 62700)

To evaluate the licensee's welding program and the program's implementation, the inspectors reviewed procedures, observed work in progress, and reviewed selected records. Observations were compared with applicable procedures, the FSAR and ASME B&PV Code Sections V and XI, 83S83 and Section IX, latest at the time of qualification.

Specific areas examined included the following: observation of welding activities in the containment associated with the replacement of the containment coolers; observation of welding activities in the auxiliary building associated with the bypass of the Boron Injection Tank, and observation of welding activities in the auxiliary building associated with the replacement of service water valves; inspection of the welding material issue station; and inspection of the welder qualification test facility. The inspectors scrutinized Work Order (WO) Nos. 11276 and 11284 for the replacement of valve Nos. V044C and V0010C.

The inspectors reviewed records for welders, Quality Control (QC) inspectors, and materials utilized in the WOs. These records included: Welding Procedure Specifications (WPS) and their supporting Procedure Qualification Records (PQR); Welder Performance Qualification (WPQ) records; records attesting to the maintenance of welder qualification; receiving inspection reports and Certified Material Test Reports (CMTR) for welding filler materials; and records attesting to QC inspectors qualification, certification, and visual acuity.

b. Observations and Findings

The inspectors noted mechanical force in the form of a chain fall was used to hold Service Water piping in place during fit-up activities prior to welding. This indicated that the pipe was flexed or cold sprung into place, thereby inducing stresses in the piping. Investigation by the inspectors revealed the following:

- Programmatically the only guidance for cold spring is contained in FNP-0-SPP-GW-002, Revision 18, "General Welding Standard For Pressure Boundary Applications," paragraph 8.6 b., which states, "There are no indications of excessive cold spring at the time of joint fit-up." The licensee had no guidance to define "indications of **excessive** cold spring".

- The pipe deflections were not documented at the time of fit-up.

- Prior to the application of the mechanical force (cold spring), no formal analysis had been conducted to determine whether the stress levels induced in the piping exceeded Code allowable levels.

- The licensee informed the inspectors that they had approved the use of cold spring at a number of locations in the SWS, because each instance did not appear visually to be "excessive." It was recommended by professional pipe fitters that pipe spring was considered by the licensee to be within the "skill of the craft" for pipe fitters.

The introduction of unknown stress levels into safety-related piping systems as shown above indicated a lack of control of the special process of welding, and is identified as an example of VIO 50-348/97-05-05: Failure to Control the Special Process of Welding. The licensee documented this issue in OR 1-97-130.

ASME B&PV Code, Section IX, requires the thickness of side bend specimens, used for the evaluation of welder qualification test assemblies, to be 3/8-inch thick. Procedure FNP-0-SPP-WP-030, Revision 15, "Specification for Welder Qualification for Pressure Boundary Applications," requires the thickness of side bend specimens to be 3/8-inch thick with no tolerance specified. An inspection of bend specimens used to evaluate welder performance test assemblies revealed several specimens that were 1/32 to 1/16-inch under 3/8-inch in thickness. Because the side bend thickness is proportional to the stress applied to the outer fibers of the bend, undersized specimens constitute a less rigorous test than intended by ASME B&PV Code Section IX. The licensee's failure to conduct bend testing on welder test assemblies in accordance the ASME B&PV Code Section IX, indicated a lack of control of the special process of welding, and is identified as an example of VIO 50-348/97-05-05. The licensee documented this issue in OR 1-97-150.

FNP-0-SPP-WF-001, Revision 12, "Procedure for Welding Filler Material Control," Paragraph No. 8.4, states, "Work areas shall be kept clear of unauthorized, unidentified or discarded welding filler materials. FNP-0-SPP-WF-001, Revision 12, "Procedure for Welding Filler Material Control," Paragraph No. 8.1, states in part, "...it is the responsibility of the welder to maintain control of filler materials until used, discarded or returned to the storeroom." Further, the licensee stated that it is their expectation that at the end of shift, work areas are free of all welding filler materials. Contrary to the above on April 11, 1997, between the night shift and the day shift, the inspectors found a significant quantity of partially used bare welding filler material abandoned in the area of the Boric Acid Injection Tank. Although some of the rods had flag tags that identified them as to type, there was no traceability to heat or batch identification. Many of the rods were of a length suitable for continued use. The licensee's failure to control welding filler materials indicated a lack of control

of the special process of welding, and is identified as an example of VIO 50-348/97-05-05. The licensee documented this issue in OR 1-97-149.

The licensee's introduction of unknown stress levels into safety-related piping systems resulting from their failure to provide guidance to define "indications of excessive cold spring;" failure to conduct bend testing on welder test assemblies in accordance the ASME B&PV Code, Section IX, resulting from their failure to provide tolerances on the bend specimen thickness; and failure to control welding filler materials resulting from their failure to adequately communicate their procedural requirements and expectations for welding filler material control to their contract welders, demonstrates less than effective control of the special process of welding and is a violation of Title 10 CFR 50, Appendix B, Criterion IX, which requires that measures be established to assure that special processes including welding be controlled. This violation is identified as VIO 50-348/97-05-05.

The licensee's Welding Manual procedures incorporates code requirements without providing guidance for implementation, some examples are as follows:

- FNP-0-SPP-GU-003, paragraph 13.1a states, "Undercut shall not exceed 0.01 inch deep when its direction is transverse to a primary tensile stress in the part that is undercut, nor more than 1/32-inch for all other situations." No guidance is provided to determine the direction of primary tensile stress.

- FNP-0-SPP-WP-030 requires the thickness of side bend specimens to be 3/8-inch thick with no tolerance specified. As discussed above, the failure to specify a tolerance caused some specimens to be made under size.

- FNP-0-SPP-GW-002 states, "There are no indications of excessive cold spring at the time of joint fit-up." As discussed above, the failure to provide quantitative guidance for cold spring resulted in the introduction of unknown stress levels into safety-related piping systems.

The Authorized Nuclear Inservice Inspector (ANII) determined that the contract welders were uninformed concerning a number of requirements of the licensee's Welding Manual and welding compliance issues. In addition structural contract welders were not provided with the means to measure base metal temperature prior to the resumption of welding, to assure compliance with interpass temperature requirements of the WPS. This issue was documented in OR 1-97-092.

A licensee QC inspector identified an instance where a contract welder was welding prior to the completion of the welder's Performance Qualification Test Record. This issue was documented in OR 1-97-083.

A licensee QC inspector identified an instance where a contract welder had made an unauthorized base material repair without benefit of a repair procedure. The contract welder burned through the material on which he was welding, and subsequently made a unilateral decision to effect a repair. The licensee viewed this decision and subsequent unauthorized repair as a coverup. This issue was documented in OR 1-97-071.

The inspectors identified a Welder Qualification Test (WQT) Record for welder 420-04-1378 on test WQT No. ST-6, Revision 2, completed on March 6, 1996, that was missing the certifying signature. The licensee indicated that they would conduct a records review to determine whether welder 420-04-1378 performed any welding on or after March 6, 1996, that was only supported by WQT No. ST-6, Revision 2 test. If so, they will take appropriate actions. The licensee subsequently determined that the test that the welder in question had taken was a licensee specific test for "T," "K," and "Y" connections. They further determined that although the welder had welded on 13 WOs since taking the test, those WOs contained no "T," "K," or "Y" connections.

The inspectors identified no discrepancies associated with welding being performed by the licensee's permanent employees.

Except as noted above the welding activities examined, were conducted by properly qualified and certified welders, using correct and certified welding filler materials in accordance with qualified Welding Procedure Specifications. Procedure Qualification Records were reviewed and determined to be adequate. Quality Control inspectors associated with the repair and replacement activities were properly qualified and certified.

c. Conclusions

A violation was identified associated with the control of the special process of welding. A weakness was identified associated with the licensee's control of contracted welders.

M1.12 Audits

a. Inspection Scope (IP 62700)

To evaluate the licensee's Audit Program as it relates to maintenance, the inspectors requested all the audits and self-assessments conducted in the maintenance area during the previous 12 months. The inspectors reviewed the two audits provided (96-STPm/34-1 Surveillance Testing - Maintenance and 96-MAINT/15-1 Maintenance Department, Routine Scheduled).

b. Observations and Findings

Audit 96-STPm/34-1 contained no findings. Audit 96-MAINT/15-1 findings included weaknesses related to: meteorological tower instruments not in agreement with the FSAR; PMT guidelines not consistently followed; smoke detector procedures performed without proper release; personal hold tag discrepancies; hand-operated hoists used with out-of-date color codes; and requirements of Purchase Orders not met. Appropriate corrective actions were taken or planned.

The audit 96-MAINT/15-1 finding related to post maintenance testing guidelines not consistently being followed, was addressed in Corrective Action Report (CAR) No. 2201, Revision 1. CAR 2001, Revision 1, did not include an adequate job of determining the extent of the problem, as only the specific examples identified by the auditors were corrected.

c. Conclusions

The area of maintenance was subjected to independent audits, with appropriate action generally taken for identified weaknesses.

M8 Miscellaneous Maintenance Issues (IP 92902)

M8.1 Painting

a. Inspection Scope (62707)

The inspectors observed painting activities, and reviewed procedures and paint data sheets from FNP-0-CP-MD-801, "Coating Program," Revision 7. Also, the inspectors interviewed various licensee personnel including supervisors, engineers, painters, and the painting foreman.

b. Observations and Findings

The licensee commenced a major painting effort throughout the plant this spring. This effort included the Auxiliary Building (AB), Turbine Building, and Diesel Generator Building. Overall the painters appear to be doing a good job. However, several problems with the EDGs and an issue with the Unit 2 PRF systems were identified as a result of the painting.

- Licensee personnel identified that the fusible links for the 1-2A and 2B DG roll up doors had been painted on one side. The licensee promptly instituted the appropriate compensatory actions per the Fire Protection Program until the links were replaced. Followup testing of the painted links identified that the paint raised the actuation point approximately 10 degrees F. The slight increase in actuation temperature was insignificant.

- On April 24, 1997, during a routine tour of the AB, the inspectors noticed very strong paint fumes from recent painting efforts of the RHR HX room (inside the penetration room boundary (PRB)). The paint was a modified epoxy-phenolic product with 56% solids and thinned with up to 1 pint of thinner per gallon of paint. One gallon of paint would cover approximately 225 ft^2 if applied at the recommended thickness of four mils. The inspectors were concerned about the potential operability effects of the volatile organic compounds (VOC) on the PRF system charcoal filters.

  The inspector discussed the control of painting inside the PRB with licensee personnel. The licensee staff stated that painting inside the PRB was limited to less than 1000 ft^2 in any 24 hour period which is about four gallons if applied per the paint data sheet. Neither the inspector nor the licensee staff were able to find this limit proceduralized to control the painting. Although the licensee did not have any direct procedural controls, the inspector determined that this criteria was being followed, based on interviews with operations personnel, the painting foreman, and various painters. This criteria was identified obliquely in the "Precautions and Limitations" sections of FNP-1/2-STP-20.0, "Penetration Room Filtration System Train A(B) Quarterly Operability and Valve Inservice Test," Revisions 25 and 15, FNP-1/2-STP-20.2, "Penetration Room Filtration System Train A(B) Monthly Operability Test," Revisions 7 and 7, and FNP-1/2-SOP-60.0, "Penetration Room Filtration System," Revisions 10 and 12. The procedures all stated "Do not run Penetration Room Filtration within 24 hours following significant painting (>1000 ft^2) in the penetration rooms."

  The licensee was not able to provide an existing analysis documenting the 1000 ft^2 limit. On May 9, the inspectors discussed the continuing painting efforts of rooms within the PRB without an analysis with senior licensee management. Licensee management stated they would stop painting in rooms served by a safety-related charcoal unit until the issue was resolved. This is identified as URI 50-364/97-05-06, Painting Effects On PRF Operability, pending review of the licensee's analysis.

- On April 25, 1997, the inspectors toured the EDG building to specifically inspect the EDG fuel racks, valves, and other components for paint related problems. The inspectors found no overspray on the fuel racks. However, some paint was found on the valve stems for the air header drains on one EDG. This condition was identified to the EDG SO and immediately corrected.

c. Conclusions

Painting was generally well-controlled, although, some minor deficiencies which did not affect operability were identified with overpainting of components in the EDG building. However, the painting of rooms in the PRB, while controlled, did not appear to have an analysis to document continued operability of the PRF charcoal filters.

M8.2 (Closed) VIO 50-348, 364/96-006-01: Failure to Perform Surveillance Test of SI Handswitch Input

(Closed) LER 50-348, 364/96-004: Surveillance Requirements Not Met for Manual Safety Injection Input into the Reactor Trip System

(Closed) LER 50-348, 364/96-004-01: Surveillance Requirements Not Met for Manual Safety Injection Input Into the Reactor Trip System

The licensee's review of Generic Letter 96-01 identified that the 18-month surveillance test for the manual SI input into the reactor trip system had not been performed since initial preoperational startup testing. The licensee implemented FNP-1/2-STP-38.3, "Verification of Manual SI Actuation Input to Reactor Trip," to provide guidance and a means to document completion of the test. The Unit 2 SI Handswitch was satisfactorily tested prior to Mode 2 entry after the Fall 96 outage. The inspectors observed the satisfactory performance of the Unit 1 SI Handswitch test on April 23, 1997. The licensee also performed a broadness review of events from January 1993 to August 1996 which resulted in an LER or were identified as a near miss. This review was documented in Corrective Action Report 2208 and concluded that the primary root cause appeared to be a failure of personnel to adequately self-check. The inspectors verified the licensee's corrective actions were appropriate and complete.

M8.3 (Closed) IFI 50-364/96-013-02: Increased Frequency Test Program for Charging Pumps Due to Cladding Cracking

This item was opened pending formal recommendations and actions for performing increased frequency testing of the charging pumps due to cladding cracks. Southern Company provided the formal recommendations for increased frequency testing to FNP by letter dated April 9 1997. The recommendations were:

- 2A CCP: UT every six months for three cycles then UT every 36 months. VT after three cycles then every five years (actually will be done every 54 months).

- All other CCPs: UT every 36 months and VT upon disassembly.

Based on a review of the available data and verification of the testing incorporated into the PM data base, the inspectors concluded these

intervals would be adequate to ensure that pump degradation would be identified prior to any operability concerns.

M8.4 (Closed) LER 50-348, 364/96-002: Technical Specifications Surveillance Requirements Not Met and Common Cause Failure Identified

This issue was discussed in detail in IR 50-348, 364/96-04 and closed out as NCV 96-04-03, Failure To Adequately Test RCP Underfrequency Reactor Trip Relays. No new issues were revealed by the LER.

III. Engineering

E1 Conduct of Engineering (IP 37551)

E1.1 Overstressed Unit 1 Reactor Coolant Loop Due To Inadequate Gaps Between SG Lateral Supports

a. Inspection Scope (37700)

The inspectors reviewed the pipe stress and support calculations, modification packages, and inspection records and discussed the SG support gap on SG loop C with the engineers from Westinghouse Electric Company, Southern Nuclear Operating Company (SNC), and Southern Company Services (SCS) to determine if licensee activities complied with industrial standards, regulatory requirements, licensee commitments, and American Society of Mechanical Engineers (ASME) and American Institute of Steel Construction (AISC) codes.

b. Observations and Findings

The licensee identified that excessive vibration had occurred on the RCP in Loop C of Unit 1 during normal operation. The licensee performed an inspection at cold shutdown condition during the current scheduled refueling outage and found that several of the lower lateral supports on reactor coolant loop (RCL) C piping had excessive gaps and that support LS-12 had a gap 0.7 inch smaller than designed. The designed gaps are established to allow for SG thermal growth to close toward, but not touch, the supports during normal operation temperature of 610 degrees F (°F). The licensee determined that the excessive gaps would cause no problem. The smaller gap would cause binding to occur in Support LS-12 when the operating temperature exceeded 445 °F, which would induce thermal expansion stress in Loop C piping, components, and supports. The thermal expansion stress could represent a situation that caused damage to the piping or supports.

The licensee took the following actions to correct the problem:

- Performed stress analyses and evaluation for the piping, components, and supports based on the ASME Code Section III and AISC Code.

- Inspected piping, components, and supports for cracks, deformations, and distress based on the ASME Code Section XI.

- Modified the existing shims by providing adjustable shims to adjust the gaps to the desired dimensions.

The inspectors reviewed the following calculations performed by Westinghouse and SCS for the licensee:

1) Westinghouse Calculation W-SMT-97-067, Farley 1 RCL Support Interference Thermal Pipestress Run, Rev. 0

2) Westinghouse Calculation W-SMT-97-066, 3D Finite Element Analysis of the Hot Leg 50 Degrees Elbow Using Jamming Loads, Rev. 0

3) Westinghouse Calculation W-SMT-97-061, Farley Unit 1 Uprating/Jamming - RCL Fatigue Evaluation, Rev. 0

4) Westinghouse Calculation CSE-04-97-0036, Analysis of Steam Generator Lower Support Strut Jamming Condition, Rev. 1

5) SCS Calculation SC-97-1-9162-001, Evaluation of LS-12 Embed, Steam Generator 1C, Rev. 0

Calculation W-SMT-97-067 used a 3D computer model and pipe stress computer program PIPESTRESS for the system analysis. The model included three loops, a reactor vessel, and three steam generators. The model also considered gaps, the local shell flexibility of the reactor vessel nozzle shell junction, and the horizontal stiffness of the reactor vessel supports. The results indicated that the critical stress was 67 kips per square inch (ksi) at Node 3216 of the elbow area, which exceeded the ASME Code allowable stress of 51 ksi.

Calculation W-SMT-97-066 used a 3D finite element program WECAN and the loads from the calculation W-SMT-97-067 to get more accurate results based on a finite element model. The critical stress was 45.3 ksi at Node 15792 at the safe end of the reactor nozzle. The critical stresses at the weld between the elbow and the SG and at the elbow were 43.2 ksi and 40.6 ksi respectively. Both stresses were within the allowable stress of 51 ksi and were acceptable.

Calculation W-SMT-97-061, performed for the fatigue analysis, was based on the past plant operating data such as start-up cycles, transients, and predicted future operations. The critical accumulated usage factor

was 0.974 at the outlet nozzle of the reactor vessel. The calculated usage factor was less than 1.0 and therefore the current piping condition is acceptable.

Calculation CSE-04-97-0036 was performed to assess the condition of the steel portion of support LS-12 utilizing a calculated direct force of 2494 kips and a shear force of 113 kips. The stress ratio for the postulated seismic event upset and faulted conditions were 1.22 and 0.98 respectively. The 1.22 ratio exceeded the allowable value of 1.0. Thus, the steel could have been overstressed during a seismic event. The licensee concluded that the support was acceptable as is, though, since no deformations or cracks were found during the licensee's walkdown inspection.

Calculation SC-97-1-9162-001 evaluated the embedded steel and concrete and determined that the concrete was acceptable. The stress in the embedded steel was 12 percent over the allowable stress for the faulted condition. However, the licensee concluded this condition was also acceptable because no cracks or deformations were found during the walkdown inspection.

The inspectors concluded that the calculations were adequate. Based on discussion with the inspectors, the licensee planned to incorporate into the calculations and plant procedures several changes as listed below:

- Add the cross reference for the qualification of the critical stress 45.3 ksi at the nozzle of the SG to Calculation W-SMT-97-066.

- Clarify the critical node numbers selected for the fatigue analysis in Calculation W-SMT-97-061, which were incorrect and did not match the stresses in the analysis. The selected stresses were correct.

- Develop a procedure for monitoring the support gaps during the subsequent heatup and cooldown.

Design Change Package DCP 97-1-9162-0-001 was reviewed for adequacy of its 10 CFR 50.59 evaluation and the shim modification. The inspectors concluded that the design change was acceptable.

c. <u>Conclusions</u>

The inspectors concluded that the calculations were acceptable based on their review of portions of the calculations. The licensee promptly evaluated the potential of past damage to the piping or supports and restored the intended conditions through the modifications.

E1.2 Licensing Basis of PRF System

As documented in IR 50-348, 364/97-04, dated April 2, 1997, the inspectors concluded that the PRF system was required to operate during post-LOCA recirculation for any size LOCA and proposed various apparent violations. During the subsequent pre-decisional enforcement conference on April 18, 1997, SNC management disagreed with the NRC inspectors and stated the PRF system was only required to operate under large break LOCA conditions (i.e., Condition IV events). Following the enforcement conference the NRC determined that several violations of NRC requirements had occurred and issued a Notice of Violation (NOV) by letter dated May 6, 1997. In this letter the NRC acknowledged that SNC did not agree with the inspectors' conclusions regarding the licensing basis for the PRF system and stated that the matter was under review by the Office of Nuclear Reactor Regulation (NRR). The issue is identified as URI 50-348, 364/97-05-07, Licensing Basis for PRF System During Post-LOCA Recirculation, pending additional review of this issue by the NRC.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) URI 50-348, 364/96-013-04: Common Tap for SG Steam Flow Transmitter and Narrow Range Water Level

(Closed) LER 50-348/96-007: IEEE-279 Requirements Not Met for Protection Channel III

By letter dated April 29, 1997, the licensee submitted a request for a "proposed alternative" to Section 4.7.3, Control and Protection System Interaction-Single Random Failure, of IEEE 279-1971 pursuant to 10 CFR 50.55a(a)(3). SNC implemented interim administrative controls which provide an acceptable alternative until necessary protection/ control system hardware changes can be implemented during the next Unit 1 and 2 refueling outages in 1998 (i.e., U1RF15 and U2RF12). The NRC has reviewed SNC's administrative controls and considers them acceptable as an interim measure until such time as it can review the licensee's request. Resident inspectors have verified implementation of the interim administrative controls, interviewed responsible operators on their knowledge of these controls, and reviewed applicable procedural requirements in FNP-0-SOP-0, "General Instructions To Operations Personnel," Revision 47. This URI and LER are closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls

a. Inspection Scope (IP 83750)

Radiological controls associated with ongoing Unit 1 (U1) Refueling Outage Number 14 (U1RF14) activities and with Unit 2 (U2) routine operations were reviewed and evaluated by the inspectors. Reviewed program areas included area postings, radioactive waste (radwaste) container labels, controls for high and locked-high radiation areas, procedural and radiation work permit (RWP) guidance and general housekeeping and cleanliness. Established controls were compared against Final Safety Analysis Report (FSAR) details and documented procedural requirements to meet applicable sections of Technical Specifications (TSs) and 10 CFR Part 20.

The inspectors made frequent tours of the radiologically controlled areas (RCAs). Radiation work permit guidance and selected survey results were reviewed and discussed with responsible Health Physics (HP) staff and supervisors. The inspectors directly observed worker and HP technician performance and discussed results of radiation and contamination surveys conducted for selected equipment and facility locations. Specific radiological controls and adequacy of surveys associated with movement of materials out of the U1 Containment equipment hatch, a U2 power entry and for a piece of loose metal retrieved from the U1 reactor cavity were reviewed and discussed in detail. Further, the inspector reviewed and discussed occurrence reports for three personnel contamination events (PCEs) associated with outage activities. For the PCEs reviewed, the inspectors evaluated and discussed licensee assumptions, dose methods and skin dose results in detail.

The inspectors discussed and reviewed "As Low as Reasonably Achievable" (ALARA) program implementation, individual worker doses, and dose expenditures associated with the following U1RF14 outage job evolutions.

- RWP 197-1435, Incore Drive Work
- RWP 197-1438, Seal Table/Thimble Cleaning
- RWP 197-1439, Gamma Metrics Detector Work
- RWP 197-1465, Lower Internals Movement
- RWP 197-1480, Scaffolding - Containment
- RWP 197-1730, Nozzle Dam Removal

b. Observations and Findings

and locked-high radiation area controls were verified to be implemented in accordance with TS requirements. Postings for radiologically controlled areas were proper and in accordance with TS or 10 CFR 20 Subpart J requirements. Containers holding radwaste, contaminated materials and equipment were labeled in accordance with 10 CFR 20.1904 requirements.

In general, workers followed proper radiological controls. Radiological controls and surveys associated with the U2 power entry to inspect containment coolers and for a piece of metal having contact dose rates of 1630 rem per hour, which was retrieved from bottom of the U1 reactor cavity, were conducted in accordance with approved procedures. However, several instances of individuals reaching across radiation control boundaries during movement of equipment into and out of the U1 equipment hatch were identified on April 7, 1997.

For individuals involved in licensed activities, year-to-date (YTD) dose estimates were within regulatory limits. The maximum total effective dose equivalent (TEDE) value reported was approximately 1510 millirem (mrem). Licensee skin dose evaluations for the PCEs reviewed were thorough and technically adequate. Assumptions and details regarding physical location, length of exposure and isotopic characteristics of particles or contamination were appropriate. All skin doses were within regulatory limits with a maximum exposure of 10.7 rem to the skin of the whole body for a worker installing service water valve parts in the U1 pipe penetration room on the 121 foot elevation.

From discussion with responsible staff and from review of planning documents and dose expenditures, the inspectors verified implementation of ALARA program activities in accordance with FNP-0-Radiation Control Procedure (RCP)-19, Pre and Post Job Planning for Work in Radiation Controlled Areas of the Plant, Revision 10, dated January 9, 1997. In particular, the inspectors reviewed and discussed planning for replacement of the gamma metrics detector and the removal and reinstallation of the reactor vessel lower internals. Significant reduction in dose expenditure was identified for tasks associated with the lower internals. The current dose expenditure of approximately 807 millirem (mrem) was reduced significantly relative to previous dose expenditures of 6095 mrem and 2179 mrem for the same tasks conducted during the U1RF8 and U2RF8 outages, respectively.

c. Conclusions

Radiological controls for routine U2 operations and U1RF14 outage activities were good. Several minor isolated instances of poor radiation control practices were identified. Personal doses were

maintained within regulatory limits. ALARA program activities were implemented effectively.

R1.2 Internal Exposure

a. Inspection Scope (IP 83750)

The inspectors discussed program guidance for monitoring and evaluating possible internal exposures, and reviewed in detail licensee results for investigative whole body counts conducted during the current Unit 1 outage.

In addition, guidance for testing and test results to ensure quality of supplied breathing air for respiratory protective equipment were reviewed and discussed.

b. Observations and Findings

As of April 11, 1997, six investigative whole-body counts associated with events which could indicate potential internal exposure during U1RF14 outage activities were conducted. The maximum uptake was approximately 1.6 derived air concentration-hours (DAC-hrs) resulting in a committed effective dose equivalent (CEDE) of 3.11 mrem. Because the doses did not exceed 10 mrem, i.e., 0.2 percent of the annual limit of intake (ALI), the resultant internal exposures were not added to the individuals' official exposure records.

The inspectors verified that the compressor systems used to supply breathing air were tested to certify Grade D air for potential use during outage activities. Breathing system air samples were collected quarterly in accordance with FNP-0-RCP-110, "Radiation Control and Protection Procedure, Sampling of Service Air to Meet Respiratory Limits," Revision. 4, and FNP-1-RCP-1112, "Operation of the Containment Breathing Air System," Revision. 12. Sample results collected in March 1997, verified that the supplied breathing air quality exceeded the established limits for Grade D air specified in the Compressed Gas Commodity Specification G7.1, 1973.

c. Conclusions

Controls for minimizing internal exposure were effective. Potential uptake of radionuclides were evaluated appropriately. Licensee tests verified that the service air compressor system supplied Grade D respirable air in accordance with 10 CFR 20, Appendix A requirements.

R2 Status of Radiological Protection Facilities and Equipment

R2.1 Tours of the Unit 1 and 2 Radiologically Controlled Areas (RCAs) (IP 71750)

During the course of the inspection period, the inspectors conducted tours of the Unit 1 and 2 auxiliary building RCAs. In general, health physics (HP) control over the RCA, and the work activities conducted within it, were good.

R5 Staff Training and Qualifications in Radiation Protection and Chemistry

R5.1 Respirator Training and Fit Testing

a. Inspection Scope (IP 83750)

The inspectors reviewed and evaluated General Employee Training (GET) provided to meet the requirements of 10 CFR Part 19, and the specific training and medical certification requirements specified by 10 CFR Part 20. The frequency of training and fit testing was compared to the guidance listed in American National Standards Institute (ANSI) Z88.2, Practices for Respiratory Protection, May 19, 1992.

Current training, fit testing and medical certification for selected contractor and licensee personnel who used or were designated to use respiratory protection equipment were reviewed and discussed with licensee representatives.

In addition, the frequency of training and fit testing was compared to the guidance listed in ANSI Z88.2, Practices for Respiratory Protection, May 19, 1992.

b. Observations and Findings

The inspectors verified that GET, respiratory protection training, and respiratory medical certifications were conducted in accordance with the requirements of 10 CFR 19.12, 10 CFR 20.1703 and licensee procedure FNP-0-RCP-101, "Use and Testing of Respiratory Protection Equipment," Revision 24. From review of training records of selected individuals listed on "Respiratory Protection Record," HP Form-257, issued during March and April 1997, the inspectors verified that persons who used respiratory protection equipment were trained and medically certified in accordance with the applicable procedures.

The frequency of verifying medical certification met requirements specified in 10 CFR 1703. However, the inspectors noted that FNP-0-RCP-101 only specified training and fit testing to be conducted every five years rather than annually as recommended by ANSI Z88.2. Licensee representatives informed the inspectors that revisions to the procedure would require annual training and fit testing at a five year

interval. Following these discussions, licensee management stated that the fit testing frequency would be reviewed.

c. Conclusions

Training and medical certifications for personnel using respiratory protective equipment were conducted in accordance with the licensee procedures and met the applicable requirements of 10 CFR Part 19 and 10 CFR Part 20.

R8 Miscellaneous RP&C Issues (IP 83750, IP 84750)

R8.1 (Closed) VIO 50-348, 364/96-10-02: Failure to Follow a March 14, 1983 Order to Implement and Maintain Commitments for Special Calibration of the Containment High Radiation Monitors (CHRMs)

Licensee STPs for the loop calibration of the CHRMs did not include in-situ calibrations using electronic signal substitution for all range decades above 10 Roentgens per hour (R/hr). From review of FNP-2-STP-227.18 and FNP-2-STP-227.19, the inspectors verified that guidance was revised to include an electronic calibration check for each decade of the required response range. The inspectors also verified from review of data for completed U2 CHRM STPs conducted in November 1996 that the required electronic calibration was completed satisfactorily. Completion of the in situ electronic calibration was scheduled for the U1 CHRMs during the current outage. This VIO is closed.

R8.2 (Closed) VIO 50-348, 364/96-10-03: Failure to Label Casks of Contaminated Resins in Accordance with 10 CFR 20.1904(a) Requirements

Licensee documents and training emphasized labeling requirements based on dose rates rather than radionuclide quantities and did not require specific information detailed in 10 CFR 20.1904. The inspectors verified that FNP-0-RCP-57, "Radioactive and Potentially Radioactive Material Handling," Revision 23, required appropriate labeling information to be provided based on specific quantity of radioactive material rather than measured dose rates. From review of licensee records the inspectors verified that management expectations regarding container labeling requirements were verbally communicated in August 1996 to site health physics (HP) personnel and formal training regarding the procedural revision was conducted during December 1996. In December 1996, HP personnel conducted walk-downs of the RCA to verify compliance with labeling requirements. In addition, licensee representatives stated that additional training regarding labeling requirements for specific situations, e.g., liquids and alpha-emitting radionuclides or liquids would be reviewed and discussed with HP staff and personnel. The inspectors toured the RCA and verified implementation of the current procedural requirements. This violation is closed.

R8.3 <u>(Closed) VIO 50-348, 364/96-10-04: Failure to Follow Procedures for Proper Personal Dosimetry Use</u>

Observations by both NRC inspectors and licensee HP staff identified individuals within established RCAs not adhering to HP manual requirements nor training guidance for use and placement of personal dosimetry, including thermoluminescent dosimeters (TLDs) and digital alarming dosimeters (DADs). From review of documents and direct observations, the inspectors verified implementation of licensee corrective actions. On August 27, 1996, a memorandum was issued from the HP staff to all supervisors detailing requirements for use of personal dosimetry. The inspectors also reviewed and discussed results of periodic HP staff dosimetry observations conducted between October 29 and November 25, 1996. By November 25, the documented error rate for use of personal dosimetry was less than one percent. On November 27, 1996 memorandum documented a request to HP personnel to increase vigilance of dosimetry use by facility personnel. During tours of the RCA, the inspectors did not identify programmatic problems associated with use of personal dosimetry. This violation is closed.

R8.4 <u>(Closed) VIO 50-348, 364/96-10-05: Failure to Have Adequate Procedures for Liquid Effluent Composite Sample Storage</u>

Licensee procedures for storing composite liquid effluent samples collected for quantification of non-gamma emitting radionuclides did not require use of standard methods such as acidification to prevent plate-out of radionuclides on the storage container. The inspectors verified that applicable procedures involving the storage of composite samples were revised to require proper acidification of the liquid samples. The inspectors reviewed and discussed results of a Chemistry Incident Report (CIR) completed to determine the effect of non-preservation of affected liquid samples. Results of the study were documented in the Section 6.3, Program Deviations of the FNP Annual Radioactive Effluent Release Report, dated April 21, 1997. Results of the study indicated that for the worst case assumptions regarding plate-out, historical doses were understated from 3 to 7 percent but were within limits specified in the Offsite Dose Calculation Manual (ODCM). Based on licensee actions and documentation, this violation is closed.

R8.5 <u>(Closed) VIO 50-348, 364/96-13-05: Failure to Follow Radiation Work Permit for Use of Proper Protective Clothing</u>

The inspectors observed personnel performing selected tasks within the RCA without use of the protective clothing specified by the respective RWPs. The inspectors reviewed and discussed the results of surveys and immediate corrective actions as documented in Occurrence Reports (ORs) 96-1002 and 96-1003. In addition, the inspectors verified that personnel were retrained in RWP adherence and that additional emphasis to RWP adherence would be stressed in general employee training. In

addition, an HP memorandum, dated October 25, 1996, to site personnel detailed the poor radiological practices observed and importance of proper radiological practices. The inspectors also reviewed and discussed initial implementation of a licensee initiative to track, trend and take more effective corrective actions regarding poor radiation worker practices. During tours of the RCA, the inspectors verified that selected tasks were conducted in accordance with established RWPs. This violation is closed.

S1 Conduct of Security and Safeguards Activities

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

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 maintenance of security systems. Security personnel activities observed during the inspection period were performed well. Site security systems were adequate to ensure physical protection of the plant. However, on April 29, 1997, the inspector observed numerous individuals in Unit 1 containment who were not displaying their security badges. Certain of these individuals were challenged by the inspector regarding their badges, all of whom had their security badges inside their outer anticontamination clothing. When questioned by the inspector, they expressed a belief that security badges were not required to be displayed in containment. FNP-0-AP-42, "Access Control," Revision 25, requires "security badges will be prominently displayed in plain view at all times." But Section 8.3 of FNP-0-AP-42 does allow wearing the badge beneath outer anticontamination clothing when working in <u>high</u> contamination areas. Although the containment was posted as a contaminated area, the inspector realized that the vast majority of Unit 1 containment floor space was clean or only slightly contaminated. Discussions were held with the Security Chief and HP Superintendent regarding plant worker misconceptions and the intent of FNP-0-AP-42. The Security Chief and HP Superintendent promptly corrected the situation. Subsequent tours by the inspectors have not identified any repeats problems.

F1 Control of Fire Protection Activities

F1.1 Welding and Grinding - Unit 1

During the week of April 7, 1997, an inspector observed numerous examples of poor fire protection practices in the Unit 1 121 foot elevation PPR. Work activities during this week were extremely intensive in the PPR, as was the quantity of tools, equipment and material brought in to support the work. Welding and grinding was

pervasive throughout the PPR as part of many of the ongoing jobs (e.g., SWS valve replacements, high head safety injection discharge line replacement and reroute). All welding and grinding observed by the inspector was accomplished by contractors. During tours of the PPR and discussions with craft personnel, the inspector ascertained the following: 1) WO's did not identify applicable Open Flame Permits; 2) Open Flame Permits were not routinely posted for each job, inclusion of many work activities on one Open Flame Permit made for confusion; 3) control of combustibles within a 35-foot area of the hot work was very poor (in one instance the inspector observed a burlap bag ignite and catch fire), 4) fire watches were tasked with ancillary duties, which in a few cases necessitated the fire watch to leave the immediate area, 5) inadequate fire extinguishers (e.g., beyond inspection date, partially discharged), 6) fire extinguishers were not in the immediate area, and difficult to locate, and 7) apparent discharging of fire extinguishers without notifying Operations or supervision. Furthermore, during several containment tours in the month of April, the inspector observed additional poor fire protection practices especially for welding and grinding conducted in elevated areas. These poor practices involved floors not swept clean within 35 feet; and inadequate or non-existent covers beneath work to collect sparks. The quantity and widespread nature of the aforementioned problems suggests a weakness in the implementation of the licensee's fire protection program for controlling open flame work. Upon notification of the findings made during the week of April 7, licensee management took prompt actions. The inspectors will review the actions taken by the licensee to correct this weakness.

F2 Status of Fire Protection Facilities and Equipment

F2.1 <u>Half-Hour Kaowool Fire Barrier</u>

Inspection Report 50-348, 364/96-09, issued November 8, 1996, documented the failure to install Appendix R required fire barriers on various electrical raceways on Unit 1 including BDE-15 (Train B charging pump power cables) and BHF-24 in room 160. The failure to install an Appendix R required fire barrier on BDE-15 was cited as an example of EEI 96-410/01013. On November 14, 1996, while touring the AB, the inspectors observed the installation of Kaowool on raceways BDE-15 and BHF-24 in room 160. The wraps consisted of a 1-inch layer of Kaowool on the sides and bottoms of the trays. A one-hour rated barrier requires two 1-inch layers on all four sides.

The inspectors identified this discrepancy to the licensee on November 14. The licensee provided a copy of Fire Protection Program Reevaluation, Amendment 5 (precedes the FSAR documented Fire Protection Program), which documented the installation and basis of "half-hour barriers" on BDE-15 and BHF-24. The inspectors reviewed FSAR Appendix 9B, Fire Protection Program, and found no mention of half-hour

barriers in it or Attachment B, 10 CFR 50 Appendix R Exemptions. FSAR Appendix 9B, Attachment B, Section 21.3 on page 9B.B-91 states that:

> "The redundant charging pump power cables are provided with a barrier (two 1-inch thick wraps of Kaowool blanket) having a fire rating greater than that of the projected fire in the following rooms in fire area 1-004: train A in rooms 161, 162, 163, and 168; train B in rooms 175, 160 and 159."

This issue was discussed further with the licensee on November 15, 26, December 9, and March 20, 1997. The licensee's planned corrective action was to modify the FSAR Appendix 9B, Attachment B, to "clarify" that these raceways were wrapped with half-hour barriers (1-inch Kaowool fire wrap). The inspectors informed the licensee that this "clarification" would misrepresent FSAR Appendix 9B, Attachment B, to read as though the "clarification" was part of the original NRC-approved exemptions.

The licensee's position was that there is a licensing basis for the half-hour barriers based upon SERs and NRC inspections pre-dating 10 CFR 50, Appendix R requirements. The inspectors disagreed because the use of half-hour barriers was not identified as a specific exemption from Appendix R requirements in FSAR Appendix 9B, Attachment B. As interim corrective action, the licensee has taken action to ensure that a one-hour roving fire watch is maintained on room 160 until this issue is resolved. This is identified as URI 50-348/97-05-08, Installation Of Half-hour Kaowool Fire Barriers Without Appendix R Exemption, pending NRR review.

F2.2 <u>(Open) IFI 50-348, 364/96-006-07: Fire Main Failures</u>

This item was opened pending metallurgical analysis of the failed piping and implementation of long term corrective actions. Southern Company Services (SCS) provided the results of the metallurgical analysis and recommendations for action via letter dated December 5, 1996. A ten-inch cast iron pipe failed due to a pressure excursion at a degraded section of pipe. A four-inch pipe failed as the result of localized exterior corrosion (due to wet insulation). SCS stated that there was little in the way effective inspection that could be performed to determine the integrity of the buried cast iron-lined pipe. However, regarding the four-inch pipe failure, they did recommend visual examination of readily accessible pipe in areas where water would tend to accumulate. As of May 7, 1997, this recommendation had not been implemented. This IFI will remain open pending resolution of necessary corrective actions.

F2.3 Inadequate Life Safety Exits from Turbine Building in Case of Evacuation

On May 1, 1997, and then again on May 2, with an industrial safety specialist, an inspector walked down the TB exits. The inspector identified numerous deficiencies with the exit signs and evacuation design scheme for the turbine building from a Life Safety perspective. Examples of these deficiencies were: 1) Missing Exit signs, 2) Burned out and/or broken Exit signs, 3) Nonvisible Exit signs, 4) Exit sign for a nonexit door, 5) Inadequate fire barriers for allowing personnel to safely exit the south end of the TB, and 6) No directions or signs in the 155-foot stairwells to guide personnel on how and where to evacuate the TB. These problems were pointed out to the safety specialist and discussed with plant management, who were evaluating necessary corrective actions.

V. Management Meetings and Other Areas

X1 Review of Updated Final Safety Analysis Report (UFSAR) 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 the following discrepancies:

(a) Table 7.3-1, FUNCTIONS INITIATED BY ENGINEERING SAFETY FEATURES ACTUATION SYSTEM. This table implies that PRF will actuate simultaneously with equipment that is started on a SI signal. Phase A and phase B actuation equipment is identified separately and later in the table and PRF is not identified as actuating equipment.

(b) Table 7.3-9, FAILURE MODE AND EFFECTS ANALYSIS, PENETRATION ROOM FILTRATION SYSTEM. This table lists the analysis concerning the effect on the system if a failure occurs and a PRF component is not automatically aligned during a Phase A CTMT isolation signal. (This represents an identification problem.)

(c) The Staff's SER, NUREG-75/034, SAFETY EVALUATION REPORT JOSEPH M. FARLEY NUCLEAR PLANT UNITS 1 AND 2, dated May 2, 1975, states that the PRF will actuate on a Safety Injection signal and this statement has not been incorporated into the FSAR or into plant design.

(d) FSAR Chapter 1.3, COMPARISON TABLES, Table 1.3.1, DESIGN COMPARISON identifies that FNP system functions are similar to those of the North Anna and Surry Nuclear Power Plants. However, with regard to the PRF system (Section 7.3 of the FSAR) the FNP is not similar.

(e) FSAR page 7.3-11, section 7.3.2.1.1, Single Failure Criteria, sentence 6 states that CTMT spray is activated on high-high containment pressure signal, however it actuates on high-high-high containment pressure.

(f) FSAR page 6.2-84, section 6.2.3.3.2 references paragraph 15.4.1.3.4, which does not exist.

Responsible licensee management were informed on each of the above discrepancies, many of which the licensee had already identified as part of its FSAR Reverification Program. Several of the discrepancies are involved with the NRC/SNC reviews regarding PRF system design and licensing basis (Section E1.2).

X2 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on May 15, 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

M. Ajluni, SNC (Corporate) Licensing Manager - Farley Project
W. Bayne, Chemistry Superintendent
C. Byrd, Southern Company Services (SCS) Support Design Engineering Manager
S. Casey, Engineering Support Supervisor - Steam Generators
R. Coleman, Maintenance Manager
J. Fridrichsen, SNC (Corporate) Senior Project Engineer
S. Fulmer, Technical Manager
J. Garlington, Nuclear Support General Manager
D. Graves, Health Physics Supervisor
D. Grissette, Operations Manager
P. Harlos, Plant Health Physicist
T. Harrison, Williams Power Corporation (WPC) Site Manager
J. Hayes, Fire Marshall
R. Hill, General Manager - FNP IP 73753: Inservice Inspection
C. Hillman, Security Chief
T. Liu, Westinghouse Electric Company (WEC) Farley Special Project Manager

G. Lofthus, NDE Level III Inspector
R. Martin, Superintendent Operations Support
A. Maze, NDE Project Supervisor
J. McGowan, SNC (Corporate) SAER Manager
M. Mitchell, Health Physics Superintendent
B. Moore, SNC (Corporate) Nuclear Support Manager
C. Nesbit, Assistant General Manager - Support
J. Odom, Superintendent Unit 1 Operations
J. Powell, Superintendent Unit 2 Operations
C. Sterzil, WEC Farley Special Project Support Design Engineer
L. Stinson, Assistant General Manager - Plant Operations
J. Thomas, Engineering Support Manager
P. Webb, Technical Training Supervisor
R. Winkler, Plant Modifications and Design (PMD) Supervisor
R. Yance, PMD Manager

NRC

J. Zimmerman, Project Manager - Farley Nuclear Plant

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 37700: Design Changes and Modifications
IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 60710: Refueling Activities
IP 61726: Surveillance Observations
IP 62700: Maintenance Implementation
IP 62707: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 73753: Inservice Inspection
IP 83750: Occupational Radiation Exposure
IP 84750: Radioactive Waste Treatment, and Effluent and Environmental Monitoring
IP 92901: Followup - Operations
IP 92902: Followup - Maintenance
IP 92903: Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
VIO	50-348, 364/97-05-01	Open	Failure to Notify NRC of Change of Licensed Operator Medical Status (Section O5.1).
IFI	50-348, 364/97-05-02	Open	Foreign Material in Containment ECCS Sumps (Section M1.7).
VIO	50-348, 364/97-05-03	Open	Failure to Follow Multiple TS Surveillance Requirements (Section M1.8).
URI	50-348, 364/97-05-04	Open	EDG 50% Load Reject Surveillance Testing (Section M1.8).
VIO	50-348/97-05-05	Open	Failure to Control the Special Process of Welding (Section M1.11).
URI	50-364/97-05-06	Open	Painting Effects on PRF Operability (Section M8.1).
URI	50-348, 364/97-05-07	Open	Licensing Basis for PRF System During Post-LOCA Recirculation (Section E1.2).
URI	50-348/97-05-08	Open	Installation of Half-hour Kaowool Fire Barriers Without Appendix R Exemption (Section F2.1).
NCV	50-348, 364/97-05-09	Open	Failure to Fully Implement Corrective Actions (Section O7.1)

Closed

Type	Item Number	Status	Description and Reference
LER	50-348/97-002	Closed	Safety-Related 4160 Volt AC Breakers Not Seismically Qualified (Section O8.1).
LER	50-348/97-004	Closed	Safety-Related 600 Volt AC Breakers Position Sensitive Seismic Qualification (Section O8.1).
LER	50-364/96-001-00	Closed	Reduction/Resumption of 2B Diesel Generator Speed (Section O8.2).

VIO	50-348, 364/96-006-01	Closed	Failure to Perform Surveillance Test of SI Handswitch Input (Section M8.2).
LER	50-348, 364/96-004	Closed	Surveillance Requirements Not Met for Manual Safety Injection Input into the Reactor Trip System (Section M8.2).
LER	50-348, 364/96-004-01	Closed	Surveillance Requirements Not Met For Manual Safety Injection Input into the Reactor Trip System (Section M8.2).
IFI	50-364/96-013-02	Closed	Increased Frequency Test Program For Charging Pumps Due to Cladding Cracking (Section M8.3).
LER	50-348, 364/96-002	Closed	Technical Specifications Surveillance Requirements Not Met and Common Cause Failure Identified (Section M8.4).
URI	50-348, 364/96-013-04	Closed	Common Tap for SG Steam Flow Transmitter and Narrow Range Water Level (Section E8.1).
LER	50-348/96-007	Closed	IEEE-279 Requirements Not Met for Protection Channel III (Section E8.1).
VIO	50-348, 364/96-10-02	Closed	Failure to Follow a March 14, 1983 Order to Implement and Maintain Commitments for Special Calibration of the Containment High Radiation Monitors (CHRMs) (Section R8.1).
VIO	50-348, 364/96-10-03	Closed	Failure to Label Casks of Contaminated Resins in Accordance with 10 CFR 20.1904(a) Requirements (Section R8.2).
VIO	50-348, 364/96-10-04	Closed	Failure to Follow Procedures for Proper Personal Dosimetry Use (Section R8.3).
VIO	50-348, 364/96-10-05	Closed	Failure to Have Adequate Procedures for Liquid Effluent Composite Sample Storage (Section R8.4).

VIO	50-348, 364/96-13-05	Closed	Failure to Follow Radiation Work Permit for Use of Proper Protective Clothing (Section R8.5).
NCV	50-348, 364/97-05-09	Closed	Failure to Fully Implement Corrective Actions (Section O7.1)

Discussed

Type	Item Number	Status	Description and Reference
IFI	50-348, 364/96-006-07	Open	Fire Main Failures (Section F2.2).