U.S. NUCLEAR REGULATORY COMMISSION (NRC)

REGION II

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Report No:	50-348/96-13 and 50-364/96-13
Licensee:	Southern Nuclear Operating Company (SNC), Inc.
Facility:	Farley Nuclear Plant (FNP), Units 1 and 2
Location:	7388 North State Highway 95 Columbia, AL 36319
Dates:	October 13 - November 23, 1996
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Approved by:	P. Skinner, Chief, Projects Branch 2 Division of Reactor Projects

Enclosure 2

EXECUTIVE SUMMARY

Farley Nuclear Power Plant, Units 1 And 2
NRC Inspection Report 50-348/96-13, 50-364/96-13

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

Operations

- Operations performed well in controlling plant conditions during Unit 1 steady state full power operation and Unit 2 shutdown. The conduct of Operations personnel and management was consistently in compliance with procedures and regulatory requirements (Section O1).

- Shift operators remained very attentive to plant conditions, and were quite knowledgeable of plant status and ongoing activities. However, the shift superintendent needs to consistently ensure that distracting activities in main control room (MCR) are kept to a minimum (Section O1.1).

- The defueling and refueling of Unit 2 was accomplished in a professional and competent manner; although, there was a considerable number of minor instances where foreign objects were found in the reactor cavity and spent fuel pool (SFP) (Section O1.2).

- Operators responded well to a Unit 2 solid plant pressure transient (Section O1.4).

- Overall housekeeping and physical conditions were generally just adequate. However, housekeeping in the Unit 2 radiologically controlled area (RCA) (especially the piping penetration rooms, decontamination room, and new fuel storage area) was considerably improved over previous outages (Section O2.1).

- Safety system walkdowns verified selected systems were properly aligned and capable of fulfilling their design function (Sections O2.2 and O2.3).

- An unresolved item was identified concerning the interpretation of technical specification (TS) requirements for penetration room filtration (PRF) system operability requirements (Section O2.6).

- Licensee efforts to identify, resolve, and prevent problems remained effective (Section O7.1).

- Conduct of Nuclear Operations Review Board met TS requirements and appeared thorough (Section O7.2).

Maintenance

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

- Unit 2 service water system code boundary valve replacement activities appeared to be conducted in a manner consistent with the licensee's quality programs, and reflected the licensee's understanding of the relative risk importance of this system (Section M1.2).

- The original scope of steam generator (SG) inspections, and subsequent expanded scope, demonstrated the licensee's apparently conservative approach to determining the structural integrity of the Unit 2 SGs (Section M1.3).

- Several major maintenance, modification, testing, and inspection activities were well planned and implemented during the Unit 2 refueling outage (Section M1.4, 5, 6, 8, and 11).

- A poor maintenance work practice resulted in the entry of foreign material into the Unit 2 reactor coolant pump seal injection system (Section M1.10).

Engineering

- Design change packages and plant modifications were developed and accomplished in an acceptable manner (Section E1.1 and 2). Engineering and the maintenance craft interfaced well during modification work (Section E1.1 and applicable M1 sections).

- An unresolved item was identified regarding a design issue associated with the Unit 1 and 2 SG common tap for steam flow and water level not meeting IEEE-279 (Section E1.3).

- The Unit 2 control rod test and evaluation program pursuant to NRC Bulletin 96-01 was comprehensive and satisfactorily verified control rod operability (Section E1.4)

- The remaining open commitments for Generic Letter 89-10 were completed satisfactorily (Section E8).

Plant Support

- In general, radiation work permit (RWP) guidance was adequate for routine RCA and the Unit 2 eleventh refueling outage (U2RF11) activities. Except for one individual, all personal exposures were less than administrative limits and were within 10 CFR Part 20 limits. A violation was identified for two examples of inadequate implementation of RWP dressout requirements (Section R1.1).

- The "As Low As Reasonably Achievable" program guidance and implementation were acceptable, with no negative trends identified (Section R1.2).

- Health Physics (HP) control over the RCA, and the work activities conducted within it, were good. Material condition and housekeeping in the Unit 2 RCA, considering ongoing outage activities, were much better than in the past (Section R2.1).

- No significant concerns were identified regarding radiation monitoring system (RMS) operability or supplied breathing air equipment (Section R2.2).

- Abnormal effluent releases have increased (Section R3).

- The HP organization and staffing provided appropriate radiation protection coverage of routine and outage job evolutions (Section R6).

- Proposed audits of refueling outage radiation protection activities were adequate; the planned use of outside auditors to assist was considered a program enhancement (Section R7).

- Security activities continued to be performed in a conscientious and capable manner assuring the physical protection of protected and vital areas (Section S1.1).

- A violation was identified for failing to search a vehicle prior to entering the protected area (Section S8.1).

- The number of outstanding fire protection system work requests was high. Corrective maintenance on degraded fire protection systems was being accomplished in a timely manner. Corrective actions have been effective in improving fire pump reliability. However, the root cause analyses of frequent automatic pre-action sprinkler system failures has not been effective. A program weakness was identified in that these sprinkler systems were not being maintained in their normal design configuration. Daily fire protection status reports were considered a positive means of identifying degraded fire protection systems and implementing the appropriate compensatory measures (Section F2.1).

- Surveillance tests of fire protection systems and features met the requirements of the Updated Final Safety Analysis Report (UFSAR) or evaluations had been provided to justify the deviations (Section F2.2).

- Fire protection program implementing procedures met the intent of the NRC guidelines and requirements. Procedure implementation and general housekeeping related to the control of combustibles within the plant were satisfactory (Section F3).

- Fire brigade organization and training met the facility's procedure requirements and performance of the fire brigade during a drill was good (Section F5).

- Coordination and oversight of the facility's fire protection program met UFSAR commitments. Responsible personnel worked together as a team, along with coordination by the Fire Marshall, to implement the site fire protection program (Section F6).

- Audits and assessments of the fire protection program were thorough with corrective actions taken on major discrepancies in a timely manner. However, resolution on recommendations and comments to enhance the fire protection program were not timely (Section F7).

- Evaluations of fire protection related Information Notices (IN) were appropriate and the required corrective actions had been completed, except for IN 93-41 and IN 95-36 (Section F8).

Report Details

Summary of Plant Status

Unit 1 operated continuously at 100% power for the entire inspection period. On November 22 the unit achieved 200 days of continuous operation.

Unit 2 remained shutdown for its eleventh refueling outage during the entire inspection period. The original 48 day refueling outage was extended to 64 days due to unexpected increase in repair scope of SG U-tubes. On October 23, the reactor core was defueled, and on November 21, the reactor core was reloaded. Restart was scheduled for December 14, 1996.

I. Operations

O1 Conduct of Operations

O1.1 Routine Observations of Control Room Operations

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

Resident inspectors and a regional inspector (during the week of November 4 - 8, 1996) conducted frequent inspections of ongoing plant operations in the MCR to verify proper staffing, operator attentiveness, adherence to approved operating procedures, communications, and command and control of operator activities. The inspectors also regularly reviewed operator logs and TS Limiting Condition of Operation (LCO) tracking sheets, walked down the MCBs, and interviewed members of the operating shift crew to verify operational safety and compliance with TS. The inspectors attended daily plant status meetings to maintain awareness of overall facility operations, maintenance activities, and recent incidents. Morning reports and Occurrence Reports (OR) were reviewed on a routine basis to assure that potential safety concerns were properly reported and resolved.

b. Observations, Findings and Conclusions

Overall control and awareness of plant conditions during the inspection period were excellent. During tours of the MCR, the inspectors observed that the Unit 1 MCBs were frequently in a "blackboard" condition. Whereas, the EPB had one persistent annunciator alarm and Unit 2 outage conditions resulted in numerous annunciator alarms. Aggressive efforts to reduce MCB deficiencies to very low levels were effective. The combined number of MCB deficiencies have been reduced to less than half the number that was existing earlier this year.

Operator attentiveness and response to plant conditions was generally very good; however, on occasion, certain distractions were observed. Although access to the MCR was regulated for Operations business only, some crews allowed personnel not assigned to the MCR to linger and discuss non-work related issues for up to 10 minutes. At times both reactor operators for Unit 1, which was at 100% power, had their backs to the MCB, engaged in non-work related conversation with the Shift Supervisor (SS) and personnel not assigned to the MCR. Interviews with the operators indicated that they were aware of plant conditions and the status of on-going activities. Even though no adverse consequences resulted from these few lapses in attentiveness, this practice may not be conducive to maintaining the record of exemplary operator response to slowly developing transients documented in previous reports during this cycle.

O1.2 Unit 2 Defueling and Refueling Operations

a. Inspection Scope (IP 60710)

Resident inspectors observed defueling activities on October 22, during the day and night shifts. The inspectors also observed refueling operations during the period November 19 through 21. Activities were observed in the MCR, SFP, and containment.

b. Observations and Findings

All defueling and refueling activities observed by the resident inspectors were performed in a well controlled and methodical manner in accordance with (IAW) FNP-2-UOP-4.1, Controlling Procedure For Refueling, and FP-APR-R11, Westinghouse Refueling Manual. The inspectors observed the refueling pre-job brief on November 15, 1996. The brief was thorough and covered the necessary information and procedural requirements. No significant incidents occurred during fuel handling and all fuel assemblies were landed in their appropriate locations. However, the licensee and an inspector identified a considerable number of minor instances of foreign materials entering the Unit 2 SFP and reactor cavity: SFP - key card, electrical pigtail, strips of adhesive tape and clear plastic, and a tie wrap; Reactor Cavity - small rubber bulb, hammer, and a three inch diameter aluminum disk. The inspector discussed the numerous foreign material intrusion problems with FNP management. Although all such materials appear to have been located and removed, responsible management was evaluating the adequacy of foreign material controls .

c. Conclusion

The inspectors concluded that fuel handling was accomplished in a professional and competent manner. Although no significant incidents occurred, there were some foreign objects found in the SFP and reactor cavity.

O1.3 Unit 2 Cooldown for U2RF11 (IP 71707)

On October 14 and 15, a resident inspector observed the cooldown and sol:d plant operation of Unit 2 IAW FNP-2-UOP-2.2, Shutdown Of Unit From Hot Standby To Cold Shutdown. Cooldown was accomplished in a controlled and purposeful manner. But, shortly after reactor coolant system (RCS) temperature was decreased below 200 degrees Fahrenheit, the licensee delayed further cooldown in order to conduct a seat leak test of the residual heat removal (RHR) loop suction valves IAW FNP-2-STP-158, RCS Pressure Isolation Valve Leak Test. Although this test was previously planned and evaluated for risk significant consequences, it did require plant conditions (i.e., isolation of RHR system) contrary to TS 3.4.10.3 for low temperature overpressure protection and certain precaution statements of UOP-2.2. Also, at this point of U2RF11, reactor core decay heat was very high. The operating crew did an excellent job establishing and maintaining unit conditions to support the seat leak tests. Appropriate TS LCOs were entered and tracked. However, the inspector became concerned that neither UOP-2.2 or STP-158 provided much, if any, guidance to the operators on the necessary plant conditions and how to control them effectively. The resident inspector discussed the lack of guidance with Operations management.

O1.4 Solid Plant Pressure Transient - Unit 2 (IP 71707)

On October 15, 1996, operators responded to an overpressure and loss of RCS inventory transient. The event was apparently caused by a charging flow perturbation when charging flow control valve (FCV) 122 went shut. An operator was very prompt in observing the decrease in charging flow and bumped open FCV-122. However, charging flow jumped to approximately 100 gpm. This increase in charging flow, in combination with the automatically reduced letdown flow, caused RCS pressure to spike and lift RHR relief valve Q2E11V0015A ("A" train) which failed to reseat. This event resulted in approximately 3000 gallons of RCS fluid being discharged to the pressurizer relief tank and required securing the reactor coolant pumps (RCP) due to low RCS pressure. The licensee conservatively secured all high voltage and low voltage switchyard work while the 2A RHR train was out of service to repair V0015A (2B emergency diesel generator (EDG) was tagged out for 18 month outage) and all penetration work.

The inspectors observed the licensee's recovery actions which included stabilizing the plant and removal and bench testing of Q2E11V0015A. The inspectors concluded: 1) the operating crew performed well in identifying the condition and stabilizing the plant; 2) the licensee staff maintained good control and took conservative actions; and 3) the investigation into the cause of the event was thorough.

O1.5 Large Service Water Spill In Unit 2 Containment (IP 71707)

On November 15, during return to service of Unit 2 containment coolers, Operations failed to prevent a 7000 gallon spill of service water (SW) into containment on the 155 foot elevation. The spill eventually flooded the containment basement level (105 foot elevation) to about six inches deep. Subsequent investigation by the licensee, as documented in occurence report (OR) 2-96-351, determined that Operations had directed contract pipefitters to loosen SW flanges to the A & B containment coolers on October 25. The loosened flanges were needed as vent paths to expedite draining both coolers. However, Operations failed to control the physical status of these containment coolers via tagorder or other applicable documentation. Consequently, when the time came to refill both coolers, the operating crew was unaware of the loosened flanges prior to reintroducing SW to the containment coolers. But even with inadequate configuration control, Operations missed an opportunity to identify the situation or minimize the spill by not walking down the coolers prior to refill or monitoring the refill operation in containment. Nobody was contaminated and no plant equipment was damaged during the spill, although SG work on 105 foot level was halted until cleanup was completed. Work on the lower level was recommenced the next day.

O2 Operational Status of Facilities and Equipment

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

General tours of FNP specific safety-related areas were performed by the resident inspectors to examine the physical conditions of plant equipment and structures, and to verify that safety systems appeared properly aligned. Limited walkdowns of a more detailed nature of the accessible portions of safety-related structures, systems and components were also performed in the following specific areas:

- Unit 1 and 2 SFP and SFP cooling systems
- Unit 2 containment
- Unit 2 turbine-driven auxiliary feedwater (TDAFW) pump rooms
- Unit 2 motor-driven auxiliary feedwater pump rooms
- Unit 1 and 2 component cooling water (CCW) pump and heat exchanger (HX) rooms
- Unit 1 and 2 hot shutdown panels
- Unit 1 and 2 vital 4160 volt alternating current switchgear rooms, trains A and B
- Unit 1 and 2 piping penetration room (PPR) on 100 foot elevation
- Unit 1 and 2 PPR on 121 foot elevation
- Unit 1 and 2 vital 125 volt direct current switchgear and battery charger rooms, trains A and B
- Unit 1 penetration room filtration system (PRF)
- Unit 1 primary sample room and radiochemistry lab
- Unit 1 and 2 RHR HX rooms

- Unit 2 RHR pump rooms
- Unit 2 containment spray pump rooms
- Unit 2 main steam (MS) valve room
- Service water intake structure (SWIS), including SW system pumps and switchgear
- Unit 1 and 2 turbine building
- Unit 2 charging pump rooms
- Unit 1 and 2 boric acid pump and mixing tank rooms

Overall material conditions and housekeeping for both units were generally adequate. Minor equipment condition and housekeeping problems identified by the inspectors were reported to the responsible SS and/or maintenance department for resolution. The physical appearance of the floor level in Unit 1 and 2 PPRs at the 121 foot elevation, and Unit 1 PPR at the 100 foot elevation, continue to look well worn with some random debris and discarded tools/material. Unit 1 was beginning to show the effects of less attention due to Unit 2 outage. However, the inspectors noticed that Unit 2 housekeeping in the PPRs looked much better than during past outages. This was a remarkable achievement when considering the SW system William Powell gate valve replacements being accomplished in the 121 foot PPR elevation. Management attention to this area was very evident. Also, the accumulation of outage solid radioactive waste (radwaste) in the decontamination room and 155 foot elevation RCA spaces and hallways (especially the new fuel storage area) was considerably improved over previous outages.

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

A regional inspector used IP 71707 to verify the operability of the following selected safety systems:

- Unit 1 CCW
- Unit 1 and 2 SFP cooling

The inspector used portions of FNP-1-SOP-23.0A, Component Cooling Water System, Revision 3, and walked down all the accessible valves and components of both trains of the Unit 1 CCW system. The inspector did not identify any immediate, safety significant problems that could adversely affect CCW system operability.

The inspector walked down accessible valves and components for both trains of the Unit 1 and Unit 2 SFP cooling system. The inspector did not identify any immediate, safety significant problems that could adversely affect SFP cooling system operability. The inspector found that overall material conditions of equipment was adequate. Some minor housekeeping, material condition, and labeling discrepancies were discussed with the licensee for correction. Examples of these discrepancies were:

- The inspector found the position indicator for valve Q1G31V007, "Demineralized Water to SFP Isolation," lying on the floor beneath the valve.

- The casing vent for the 2A SFP pump was leaking excessively despite the isolation valve being closed and the line capped. There was a collection bag under the line, and the area immediately around the pump was designated as a contamination area. However, the bag was full and leaking on the pump housing.

- The hand wheel on the skimmer pump casing drain valve, Q2G31V039, appeared to have been stepped on, bending the valve stem.

- An unauthorized operator aid was noted above the Unit 2 SFP to refueling water storage tank (RWST) isolation valve Q2G31V021B. "Open 1 1/2 turns for 100 gpm" was written in pencil on the wall.

- The sample line below Unit 1 SFP cooling sample valve, Q1G31V011A was plugged with a boron buildup.

- An inactive red hold tag was found on the SFP purification outlet to refueling cavity (N1G31V021A). (See paragraph O2.5)

O2.3 Engineered Safeguards Feature System Walkdown

a. Inspection Scope (IP 71707)

The inspector performed a detailed walkdown of the accessible portions of the Unit 1 and Unit 2 hydrogen recombiners and the Unit 1 and Unit 2 post accident hydrogen analyzers (PAHA). The inspector also used portions of FNP-1-EEP-1, Loss of Reactor or Secondary Coolant, Revision 15, in order to walkdown the equipment as it would be used in an accident situation.

b. Observations, Findings and Conclusions

The inspector found that overall material conditions of equipment was adequate with the exception of a concern with the 2A hydrogen recombiner. A conduit leading to the 2A hydrogen recombiner had pulled away from the connector exposing the wires within. This could affect the operability of the recombiners if the wires were exposed to post-loss of coolant accident atmospheric conditions. Some minor housekeeping, material condition, and labeling discrepancies were discussed with the licensee for correction. Examples of these discrepancies were:

- pencil labels on the Unit 2 PAHAs.

- no warning signs on the Unit 1 hydrogen sample lines to alert personnel to the exposed heat tracing as on Unit 2.

Other than the broken conduit, none of these discrepancies were significant enough to adversely affect the operability or operation of hydrogen recombiners and PAHA equipment.

O2.4 Containment Tours - Unit 2 (IP 71707)

Resident inspectors toured Unit 2 containment on several occasions during the outage. The initial tour, conducted two days after the unit shutdown, only identified one oil spot from a snubber leak. No RCS or other fluid system leaks were identified. However, the inspectors were concerned about trash and loose tools/equipment which was already accumulating in containment. This concern was discussed with plant management at the exit meeting for Inspection Report (IR) 50-348, 364/96-09.

O2.5 Tag Orders (IP 71707)

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

- TO# 96-1911-2: 2B CCW HX
- TO# 96-1971-1: 1B CCW HX
- TO# 94-0791: SFP purification
- TO# 96-2911-2: 2B RHR pump

All tags and TOs examined by the inspectors were properly executed and implemented, with certain exceptions. During the walkdown of valves associated with Unit 1 SFP cooling a red hold tag associated with TO #94-0791 was observed on SFP purification outlet to refueling cavity isolation valve (N1G31V021A). Review of the Unit 1 TO log revealed that this TO was no longer active. The tag had been initialed as cleared on March 20, 1994. Due to the large amount of time since the TO had been cleared, it could not be determined if the tagging official had ensured all tags had been removed.

After determining that the tag was hanging in error, the operating crew removed the tag from the valve handwheel. The valve was in its required position. OR 1-96-338 adequately addressed this issue. The inspector verified that none of the other tags associated with this TO were still hanging.

Also during the inspection period two ORs (2-96-308 and 2-96-375) were written regarding the loss of control over a 2B SG manhole cover hold tag and a nitrogen hose hold tag and caution tag on tygon hose running through penetration 90. Licensee and contractor investigations were in progress.

O2.6 Technical Specifications Compliance (IP 71707)

A resident inspector, while performing MCR observation on November 19, 1996, overheard operators discussing a possible TS compliance issue which occurred on October 30, 1996. The inspector discussed the issue with the operators and reviewed the Unit 2 MCR log entries for October 30.

On October 30, the licensee was moving fuel in the SFP. The 2A startup transformer (normal power supply to the "A" train safety busses) and the 1-2A EDG (emergency power supply for "A" train safety busses) were out of service (OOS) for outage work. At approximately 10:22 am, the licensee commenced performance of FNP-2-STP-20.2, Penetration Room Filtration System Train A (B) Monthly Operability Test, on "B" train. The lineup for this surveillance test procedure (STP) required isolating the "B" train PRF system suction from the SFP. AT this time, the "A" train PRF was running on its alternate power supply (i.e., 2B startup transformer). The night shift operators reviewed TS 3.9.13 and 3.7.8, and at 7:59 pm decided that they did not meet the TS requirements to move loads over the SFP. The operators informed the SS at 8:20 pm. The movement of loads over the SFP was terminated and Operations management was notified. The Operations manager reviewed the situation and concluded TS requirements were met. The handling of loads over the SFP was recommenced at 9:23 pm.

The inspectors independently reviewed TS 3.9.13 and 3.7.8 and determined that train A PRF was inoperable from October 29 through November 2 while the "A" train normal and emergency power supplies were OOS. Furthermore, on October 30, the "B" train PRF to the SFP was rendered inoperable by shutting the "B" train PRF suction to perform STP-20.2. The inspectors met with plant management to discuss the issue, and participated in various conference calls with the SNC corporate office. After considerable discussion, licensee management continued to disagree with the inspector's and operating crew's interpretation of TS requirements. To resolve this issue, the licensee subsequently documented their position in a letter to the NRC dated November 27, 1996, requesting a formal TS interpretation. Until the NRC responds to the SNC letter, this issue is identified as Unresolved Item (URI) 50-364/96-13-01, PRF Operability Requirements for SFP.

O7 Quality Assurance in Operations

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

The resident inspectors scanned all ORs initiated, and approved by the operations manager during the inspection period to ensure that plant incidents that effect or could potentially effect safety were properly documented and processed IAW FNP-0-AP-30, "Preparation and Processing of

Incident Reports ..." Certain selected ORs were reviewed in detail as part of the routine inspection program.

Overall, the inspectors concluded the licensee's program for identifying and resolving problems remained effective, and was being accomplished IAW AP-30. Plant personnel and management exhibited an appropriate threshold for identifying problems, initiating ORs, and assigning formal root cause teams. Each new OR received prompt attention and was regularly discussed by management in the morning status/plan of the day meeting. Direct derivations and formal root cause analyses continued to be conducted by experienced plant staff in a rigorous and thorough manner. The results of these efforts were almost always effective at preventing recurrent problems.

The following ORs were reviewed and corrective actions were verified:

- 2-96-301, MOV 3318A found open
- 2-96-346, 2A RHR motor wouldn't rotate while bumping for rotations (refer to paragraph M1.6 for more details)
- 2-96-309, Wiring discrepancies during solid state protection system walkdown

O7.2 Nuclear Operations Review Board (NORB)(IP 40500)

TS 6.5.2 defines the function, composition, responsibilities, and authority of NORB. A resident inspector monitored the NORB conducted on November 15, 1996. The inspector observed good discussions on problem areas. The inspector also verified that the NORB met TS requirements for members.

O8 Miscellaneous Operations Issues (IP 92901)

O8.1 (Closed) Licensee Event Report (LER) 348/95-10, Actuation of Engineered Safety Feature Equipment Due to Loss of Main Feedwater

This event was discussed in IR 50-348, 364/95-19. No new issues were revealed by the LER.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

Inspectors observed and reviewed portions of various licensee corrective and preventative maintenance activities, and witnessed routine surveillance testing, to determine conformance with plant procedures, work instructions, industry codes and standards, TS and regulatory requirements.

a. Inspection Scope (IP 61726, 62703 and 62707)

The resident inspectors and a regional inspector observed all or portions of the following maintenance and surveillance activities, as identified by their associated work order (WO), work authorization (WA), or STP:

- FNP-1-STP-23.1: 1A CCW Pump Quarterly Inservice Test (IST)
- FNP-0-STP-80.17: Diesel Generator 2C Operability Test
- FNP-2-STP-40.3: Phase A Isolation Test
- FNP-2-STP-627: Local Leak Rate Test of Containment Penetrations
- WA# 120102: Unit 2 Fuel Oxide Measurement
- FNP-2-STP-40.1: B2F Sequencer Operability Test and B2F, B2H Sequencer Load Shedding Circuit Test
- FNP-2-ETP-4411: 2B Residual Heat Removal/Low Head Safety Injection Pump Curve Development Test
- FNP-2-STP-228.1: Nuclear Instrumentation System Source Range N31 Calibration and Functional Test
- WO# M96004427: 2A Charging Pump Seal Repair
- FNP-2-STP-40.7: Emergency Core Cooling System Branch Line Flow Test
- WO#00079621: Relug 2B RHR Pump

b. Observations, Findings and Conclusions

All of the aforementioned maintenance work and surveillance testing observed by the inspectors were performed IAW work instructions, procedures, and applicable clearance controls. No adverse findings were identified. Safety-related maintenance and surveillance testing evolutions were well planned and executed. Responsible personnel demonstrated familiarity with administrative and radiological controls. Surveillance tests of safety-related equipment were consistently performed in a deliberate step-by-step manner by personnel in close communication with the MCR. Overall, craftsmen and technicians appeared knowledgeable, experienced, and well trained for the tasks they performed.

In addition, see the discussions below regarding certain major maintenance and testing activities observed by the resident inspectors and a Region II inspector (Sections M1.2 through M1.11).

M1.2 Service Water Valve Replacement, Unit 2 (IP 73753)

a. Inspection Scope

The inspector reviewed SW System valve replacement activities through inspection of materials; review of procedures and drawings; observation of work activities; and discussions with craft and engineering personnel. The activities were inspected for compliance with the Farley UFSAR and licensee quality requirements. This area of maintenance work was selected for observation because of the relative importance of the SW system in the licensee's IPE.

b. Observations and Findings

The licensee was in the process of replacing carbon steel gate valves with stainless steel butterfly valves as the code boundary valves between the American Society of Mechanical Engineers Class 2 and Class 3 portions of the SW system. The change was being made due to corrosion problems with the carbon steel gate valves. At the time of the inspection, the valves had been replaced in the train A portions of the SW system, and work was in progress for the train B valve replacement. In train B, the gate valves had been removed and the piping was being prepared for installation of piping and flanges necessary for the installation of the butterfly valves.

Along with a licensee materials engineer, the inspector conducted an inspection of the condition of inside surfaces of piping in the vicinity of the valve replacement locations. The piping contained an oxidized coating, several millimeters in thickness, that obscured the inside surface of the piping. The materials engineer was able to easily remove the coating with a putty knife so that various locations of the inside surface could be examined.

The piping inside surfaces were found to contain some small pitting indications, but in general the piping did not appear to have suffered appreciable wall loss due to corrosion. The ends of the piping which had been machined in place, in preparation for welding, were examined to assess the relative depth of the pitting. On the surfaces examined, the pits appeared to be only a few millimeters in depth, and therefore within the corrosion allowance for this piping wall thickness.

During discussions with craft personnel, the inspector was informed that the piping sections examined by the inspector were representative of the piping conditions noted during the entire modification project. Several of the craft personnel noted that they had expected to find the piping in poor condition, and had been impressed with it's relatively good condition, and how easily the pipe ends cleaned up for weld preparation. The inspector did witness the liquid penetrant examination of a weld repair in the Class 2 weld preparation for a 6-inch valve (Q2P16V044B). The area being repaired was due to removal of a linear indication, resulting in a 2.5-inch by 7/8-inch area being ground in the end of the pipe.

The inspector reviewed two work packages representative of the work being done; one package for a 12-inch diameter valve and the other for a 6-inch diameter valve. The inspector noted that the work packages were set up in a "traveler" format with individual work sheets for each of the different welding operations required for each valve replacement. The work packages appeared to be very complex, but after discussing the jobs with the craft personnel on the scene, it was apparent that they understood the process and how the job was to be documented.

c. Conclusions

The SW code boundary valve replacement activities appeared to be conducted in a manner consistent with the licensee's quality programs. The fact that the licensee had a materials engineer assigned to oversee these replacement activities, along with other work on the SW system, was a positive indication that the licensee understood the relative IPE importance of the system.

M1.3 SG Inservice Inspection (ISI), Unit 2 (IP 73753)

a. Inspection Scope

The inspector reviewed ISI inspection activities involving the Unit 2 SG tubing. The review consisted of discussions with licensee and contractor personnel; review of eddy current and ultrasonic test results; and an independent review of a portion of the eddy current, bobbin inspection, test data from the "C" SG.

b. Observations and Findings

Farley 2 is a Westinghouse 3-loop unit with series 51 SGs. Each SG contains 3388 U-bend tubes made of Inconel 600. The nominal tube outside diameter is 0.875 inch with a nominal wall thickness of 0.050 inch; the tubes were expanded into the tubesheet using a mechanical hardroll process. Unit 2 reached initial criticality in May 1981, and this outage is the eleventh (11th) refueling outage.

During this refueling outage, the licensee's eddy current plans included a 100% bobbin coil inspection of all tubes in each SG from the cold leg side concurrent with a 100% rotating coil inspection of the hot leg tube sheet. These basic inspections were to be followed by rotating coil inspections of bobbin coil indications, particularly at hot-leg support plate intersections and in free spans. The licensee was using the plus-point probe for the rotating coil inspection for the first time.

As a result of the planned inspections, a significant number of new indications were detected. Because of these new indications, the licensee initiated a special program to establish structural integrity of the "A" SG; this program would then be followed by special programs on the "B" and "C" SGs. There were five types of inspection findings that were determined to be "Free Span Structural Integrity Issues" which were the focus of the continuing inspection activity in the "A" SG. These findings are as follows:

Inside Diameter (ID) Circ crack:	>180° at or above the Bottom of Roll Transition (BRT)
Outside Diameter Circ crack:	>180° at or above the BRT
Axial cracks:	Crack length of >0.4 inches, above the BRT.
Axial & Circ Mixed:	If no "null" between circ and axial, Indication at or above BRT.

Free span above roll transition: Plus point indication >2 volts

In addition to these tubes, 20 Single Axial Indications and Multiple Axial Indications with axial crack length >0.3 inches, above the BRT were included in the continuing inspections.

There was a total number of 59 tubes included in the additional inspection activities. The inspection activities included: a repeat of the plus-point inspection using a slower speed in order to generate more data points, and ultrasonic examination (UTEC) of all ID cracks in the sample (52 of the 59 tubes had ID cracks).

The data from the additional plus-point and UTECs were currently being analyzed to select the appropriate sample tubes for in-situ pressure testing and also for tube pulls.

At the time of this inspection, in the "B" SG, fifteen (15) tubes had been selected for "slow" plus-point and UTEC. UTEC equipment was being installed in the "B" SG. In the "C" SG, data analysis from the original bobbin and plus-point examinations had just been completed.

One finding of note, in the "C" SG, was a bobbin coil indication at the first hot leg support plate on tube No. R34C53. This tube was plugged in October 1990 with an indication that had been confirmed by Rotating Pancake Coil (RPC). (This was before the licensee had received approval to use a voltage-based alternate plugging criteria, and tubes were plugged upon confirmation of an indication by RPC.) The tube remained plugged until March 95 when it was unplugged and bobbin coil examination showed the indication to be 1.89 volts, which was below the plugging criteria of 2 volts. During this inspection, bobbin coil examination measured the indication as 6.73 volts. This is a significant growth rate for one fuel cycle.

In discussions with the licensee the inspector learned that, while this one tube (R34C53) is a singular case because of its large apparent growth rate, there does appear to be a pattern where tubes that have been plugged and later returned to service apparently show larger indication growth rates than tubes that have remained in service.

Using the contractor's evaluation equipment, the inspector reviewed bobbin-coil, eddy current data for the following "C" SG tubes:

R34C37; R06C26; R07C35; R46C51; R45C46; R34C53;
R42C54; R45C56; R35C60; R31C60; R27C68; R30C67;
R32C65; R37C64; R41C64; R42C64; R39C65; R40C67;
R33C71; R30C72; R35C74; R35C76; R32C42; R24C78;
R26C77; R23C75.

During this review, the inspector paid particular attention to indications at the hot leg support plate locations.

The inspector also observed preparations for the UTEC of a sample of "B" SG tube indications. The preparation work activities were observed via video monitoring equipment in the contractor's trailer. The UTEC system was being used to quantify the lengths and depths of a representative sample of crack indications found using the plus-point eddy current probe. The inspector also reviewed data plots of the nine tubes from the "A" SG that had been inspected. At the time of the inspection the only indication that had been depth sized by UTEC was a half-inch long, axial indication located five inches above the top of the tubesheet in tube R28C26; that indication had been sized as less than 0.020" in depth.

c. Conclusions

Based on the review of the planned scope of the SG inspection, and the expanded scope of inspections after significant numbers of indications were found, the licensee appears to have taken a conservative approach to determining the structural integrity of the Unit 2 SGs.

M1.4 2B EDG 18 month inspection (IP 62707)

The inspector reviewed the completed work packages and observed limited portions of the following work on the 2B EDG:

- MP-14.1 18 Month Inspection
- Replacement of Lube Oil HX, Intercooler HX, Jacket Water HX tube bundles due to inlet tube sheet erosion concerns.
- Inspection/replacement of #2 Air Start Header air start check valves for exhaust backleakage.

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

M1.5 WOs #96002615, 96002618, and 96002619: Inspection of Charging Pump Casings (IP 62707)

The resident inspectors reviewed the licensee's inspection of charging pump casings for cracks and boric acid corrosion concerns identified by NRC Information Notice (IN) 94-63. The licensee utilized an ultrasonic testing (UT) inspection process to inspect the nozzle ends. This process was tested using mockups to verify it could detect defects between the cladding and the case as small as 0.09 inches. The inspectors reviewed the test package and determined it was adequate. The inspections revealed no indications of cladding cracks in the nozzle ends.

However, just prior to the outage, a failure of the 2A charging pump required removal of the rotating assembly. With the rotating assembly removed, the licensee identified rust stains on the casing clad. UT and RT testing in areas of rust stain showed no indications of a crack or wastage of the casing. The licensee videotaped the indications and sent copies to Westinghouse and Pacific Pumps for further analysis. The analysis indicated that there were possibly two cracks in the cladding but there was negligible wastage of the casing. Initial recommendations were to conduct UT inspections of the casing and nozzles on an increased frequency. However, as of the end of this report, no formal recommendations or actions had been taken. This will be tracked as Inspector Followup Item (IFI) 50-364/96-13-02, Increased Frequency Test Program for Charging Pumps due to Cladding Cracking.

M1.6 Design Change Package (DCP) S95-2-8966: Changeout of RHR Pump Impellers (IP 62707)

The resident inspector reviewed the DCP and observed work in progress and the post-modification testing. This DCP was performed to enhance pump performance at higher flows to provide a larger margin for pump degradation. The DCP was performed on Unit 1 during the last refueling outage. Overall work was well controlled. However, rework was required to relug the motor leads because an incorrect lug/crimp size was used and the motor ratings required upgrading from 380 to 400 horsepower.

The lugging deficiency was identified on November 11, 1996, when the 2A RHR pump failed to start during the post-modification testing. On investigation the licensee found an open circuit where the lug on the motor lead had pulled off the wire. The motor leads were 49 strand #6 wires. The original lugs were #4 lugs. Maintenance personnel initially tried to relug the motor leads using #6 lugs but they were too small. Maintenance personnel reverted back to the #4 lugs (as used by the manufacturer) and crimped the lugs with a #8 die. The maintenance personnel found that the lugs were not tight enough on the 2B RHR pump so they recrimped the 2B RHR pump motor leads with a #7 die. They did not go back and recrimp the lugs on the 2A RHR pump.

The licensee relugged both RHR pumps using #6 lugs specifically sized for 49 strand wire per WO S00079621. An inspector observed the relugging of the 2B RHR pump. The inspector reviewed FNP-0-EMP-1370.01, Cable Termination, Splicing and Repair, for specific guidance on lugging criteria. The inspector determined that the procedure only provided general guidance and relied on skill-of-the-craft for determining lug sizing requirements.

The inspector observed the post-modification testing of the pumps per FNP-2-ETP-4411, 2B Residual Heat Removal/Low Head Safety Injection Pump Curve Development Test, Revision 0. The inspector reviewed the test prerequisites and procedure and observed the evolution prebrief and performance of the test. The licensee found that the maximum brake horsepower (Bhp) requirement at a runout condition of 4400 gallons per minute was higher than expected (between 380 and 390 Bhp). The RHR motors have a 380 Bhp rating. To prevent exceeding the motor ratings the licensee placed administrative controls on the use of the RHR pumps until formal evaluations could be completed. The licensee performed a 50.59 evaluation, performed additional testing on the RHR pumps, and analyzed the impact of the increased loading on the EDGs. As a result of the 50.59 evaluation and the related safety evaluation, the licensee was able to uprate the RHR pump motors to 400 Bhp. The inspectors observed the Plant Operations Review Committee meeting at which the issue was discussed and reviewed the 50.59 package. The inspectors concluded the evaluation was thorough and the motor uprate was adequately justified.

M1.7 FNP-1-STP-24.1; Service Water System Quarterly IST

a. Inspection Scope (IP 61726)

An inspector observed the entire performance of FNP-1-STP-24.1, 1A, 1B, and 1C SW Pump Quarterly IST.

b. Observations and Findings

The SW sytem pumps performed as expected. The inspector verified that: 1) all initial conditions and prerequisites were satisfied and 2) test instrumentation was calibrated and of the proper range. The inspector also verified selected data point calculations.

c. Conclusions

This IST of the unit 1 Train A SW system pumps, including the swing 1C pump, was performed IAW the procedure steps of STP-24.1. No deficiencies were identified.

M1.8 STP-40.0; Safety Injection With Loss Of Offsite Power Test - Unit 2 (IP 61726)

On November 16, a resident inspector observed the conduct of FNP-2-STP-40.0, Safety Injection With Loss Of Offsite Power Test. Overall, this fully integrated, challenging and complex test was very well orchestrated. No significant procedural problems or performance errors were identified. All systems and components operated per design during the test except: 1) minipurge dampers would not reopen; 2) hydraulic contol valve 603A failed to fully open (due to binding); 3) 2A and 2B RHR pump flows were less than required; and 4) the mini-flow for 2C charging pump sprung a serious packing leak. Also, the 2A charging pump was unavailable for the test due to repairs, and the 2D containment cooler fast speed breaker was put in test just prior to STP-40.0 due to its fan rotating in reverse direction. Deficiency Reports were written to address identified equipment problems. 2A and 2B RHR flow discharge valve stops were subsequently readjusted and pump flow retested satisfactorily.

M1.9 WO# S96000458; Unit 2 TDAFW Pump Governor Valve Stem and Spacer/Washer Replacement (IP 62707)

On October 28, a resident inspector observed implementation of modification DCP-2-95-8939 by several mechanics. This DCP replaced the Unit 2 TDAFW pump governor valve stem, spacers, and washers based on problems described by IN 94-66. The concern involved the use of incompatible materials that would cause the stem to bind resulting in turbine overspeed trips. The DCP and WO directed the installation of new components made of vendor recommended materials. The inspector observed the component replacements per approved work instructions and independently verified material compositions by reviewing applicable purchase orders.

M1.10 RCP Seal Injection System Foreign Material Intrusion (IP 62707)

OR 2-96-325 was written to document an investigation into the source and cause of foreign debris (i.e., pulverized O-ring material) discovered in six of nine RCP seal injection check valves. The seal injection check valve internals were inspected during U2RF11 due to prior evidence of debris from seal injection filter O-ring material (Maintenance Incident Report 95-02) and a seal injection flow transient during Unit 2 fuel Cycle 11 (Incident Report 2-96-161). The discovery of additional O-ring material downstream of the seal injection filters led to the preliminary conclusion that a poor work practice used during installation of RCP seal injection system filters (i.e., failure to lubricate O-rings per manufacturer recommendations) has resulted in the introduction of O-ring fragments throughout the seal injection flow path to the RCP seals. The inspectors will review the licensee's completed OR and applicable safety evaluation, and verify corrective actions. This issue is identified as

IFI 50-348, 364/96-13-03, Foreign Material From Seal Injection System To RCP Seals.

M1.11 Reset Motor-Operated Valve (MOV) Torque Switches to Higher Values (IP 62707)

The NRC inspectors reviewed documentation and observed work in progress to verify that the licensee was resetting MOVs IAW a previous commitment made to the NRC in response to IR 94-28. The schedule, provided in a March 3, 1995, letter from the licensee to the NRC, required 10 Unit 2 MOVs to be reset in the current Unit 2 refueling outage. The related work observed by the inspectors involved resetting MOV 2-3210B. The documentation reviewed was selected from a sample of 3 of the 10 valves and included: work authorization 451260 for resetting MOV 2-8887B and WOs 75880 and 75865 involving activities to replace MOVs 2-3019B and 2-3134 (requiring them to be reset). The inspectors concluded that the Unit 2 MOV were being reset IAW the licensee's commitment.

M8 Miscellaneous Maintenance Issues (IP 92902)

M8.1 (Closed) LER 50-364/95-08: Reactor Trip During DEH Card Changeout

This event was discussed in IR 50-348, 364/95-20. No new issues were revealed by the LER.

III. Engineering

E1 Conduct of Engineering

E1.1 Design Changes and Plant Modifications

a. Inspection Scope (IP 37550)

The inspector reviewed design changes and plant modifications that were being implemented on Unit 2 during the current refueling outage to determine if these activities were being performed IAW regulatory requirements, licensee commitments, and the design modification procedure. Walkdowns were also performed in order to inspect the implementation of the modification in the field.

b. Observations and Findings

The licensee had 46 DCPs on the list of plant modifications that had been approved and scheduled to be worked during the U2RF11. The inspector reviewed four DCPs as listed below that involved significant electrical modifications:

DCP 95-0-8816, Provided design to convert the MCR air conditioner units from water to air cooled units. This modification is being implemented by the licensee to improve reliability, maintenance and operation of the MCR air conditioning system.

DCP 95-2-8875, Provided design to delete 13 MCR recorders and associated wiring on Unit 2, replace four recorders with computer points, replace three recorders with vertical scale indicators, add a new recorder to monitor SG feedwater pump suction pressure, replace two obsolete MCB trend recorders, and add jacks to the moveable incore detector system to provide for the connection of a portable recorder.

DCP 91-2-7186, Provided design to install interposing relays in the circuitry for high energy pipe break switches. The current high energy pipe break switches are obsolete and the approved replacement switches have only one set of contacts instead of two as on the existing switches. The addition of these relays will allow replacement of existing switches.

DCP 87-2-4592, Provided design to automatically load an instrument air compressor on the EDGs by the engineered safety system or loss of offsite power (LOSP) sequencer, block the pressurizer heater backup group A during sequencing and provide a manual bypass for unblocking.

The plant modifications required to be performed by the above DCPs had not been completed. The inspector conducted interviews with the appropriate assigned engineers to discuss the scope of the modification, work completed and remaining, and testing that would be performed for functional acceptance. The inspector, accompanied by the assigned engineer, performed walkdowns in the field to examine the work completed on modification DCPs 95-0-8816, 95-2-8875, and 91-2-7186.

With regard to DCP 95-0-8816, the inspector examined the modifications of the A and B trains of the MCR air conditioner units. The A train modifications were complete and functional. The B train was being worked this outage. The inspector examined the conduit routing for the new 600 VAC power cable for the B train condenser unit and found it to be acceptable. The routing was examined from motor control center 1G in the auxiliary building to the B train condenser unit on top of the control building. Intermediate routes also included the MCR via conduit. A review of the UFSAR and RG 1.75, Revision 0, confirmed that this routing was consistent with the licensee's commitments on RG 1.75. The inspector concluded that the modifications performed to date appeared to be acceptable.

The inspector examined portions of work performed under DCP 95-2-8875 to delete specific MCR recorders. The inspector accompanied by the assigned engineer, toured the MCR to examine the Unit 2 MCB and the recorders that were being removed. While in the MCR the inspector observed the craft determinating the Metal Impact Monitoring System Recorder. The craft noted that the wires could not be determinated at both ends because one end was soldered. The craft notified the engineer who was accompanying the inspector at that time of this problem. A short time later, the engineer issued a field change for the wires to be determinated at the one end only, taped and spared. On a subsequent tour of the MCR, the inspector observed that the Metal Impact Monitoring Recorder had been removed as required by the DCP and the internal wiring had been determinated at one end only, the ends of the wires had been sealed with tape, and each wire had been labeled spare. The inspector found the completed work to be acceptable. In regards to this issue, the inspector found that the craft had a good questioning attitude and that engineering interfaced well with the craft to resolve this problem.

The inspector held discussions with the licensee regarding the instruments that are within the scope of RG 1.97 i.e., Types A, B, and C and Categories 1 and 2 instruments. These instruments are required by RG 1.97 to be uniquely identified on MCBs as post accident monitoring instruments. The results of these discussions revealed the following information regarding the licensee's methods for labeling MCR instruments. An internal Alabama Power Company letter dated December 14, 1987, "Marking of Main Control Room Indicators," indicated that orange labels were being used as markers of EQ indicators to aide the operators in identifying those instruments that would be more reliable under accident conditions. Some time later the orange labels were removed by the licensee and replaced with nameplates with black lettering and white backgrounds with "EQ" in the label description. The licensee's RG 1.97 Compliance Review Report No. A-204866, Revision 4, dated November 28, 1995, requires that certain instruments be marked as RG 1.97, but it does not specify how these instruments will be marked or labeled. The inspector selected four Category 1 indicators from the Compliance Review Report (i.e., PI-402A, 402B, 403A, and 403B) and confirmed in the MCR that they had been marked as "EQ". The inspector also noted that DCP 95-2-8875 provides design to delete the Boric Acid Flow Strip Chart Recorder which is Variable 102 in the RG 1.97 Compliance Review Report. In accordance with the Compliance Report this variable is not required to be marked on the MCB. Although no RG 1.97 indicators were identified without labels, the inspector had a concern that the labeling of EQ instruments may not be adequate because there may be other RG 1.97 instruments that are not EQ that are required to be labeled.

The licensee acknowledged the inspector's concern regarding RG 1.97 labelling and subsequently conducted a thorough re-review of the issue. Based on this re-review, SNC issued a letter dated December 5, 1996 to the NRC revising their commitment from using orange bars on the MCBs to the "EQ" designator. Furthermore, this letter confirmed that all RG 1.97 required variables specified in their previous commitment were marked "EQ," except for MS line pressure, RWST level, and condensate storage tank level. But since these instruments were not located in harsh environments and are uniquely identified by the plant's emergency response procedures no additional identification is warranted.

The inspector reviewed portions of Revisions 7 and 8 to Design Calculation E-42, Steady State Diesel Generator Loading Calculation for LOSP, Safety Injection (SI), and Station Blackout (SBO), to confirm that the additional loads placed on the EDGs by DCPs 8816 and 4592 had been properly assessed. The inspector found that Revision 8 assessed the load additions from DCPs 8816 and 4592 and concluded that EDG 1C steady state load would exceed the continuous rating by less than 5 percent in some design basis scenarios and SBO scenarios, but would remain well below the yearly 2000 hour rating of 3100 kilowatts. This was found to be acceptable. The steady state loading for EDGs 1-2A, 1B and 2B remained below the continuous rating for all design basis and SBO events.

Field work on DCP 91-2-7186 was inspected and found to be acceptable.

The inspector reviewed the 50.59 evaluations for the DCPs identified above to verify that they were adequate and that an unreviewed safety question did not exist.

c. <u>Conclusion</u>

The inspector concluded that the design changes and plant modifications were adequate and the work completed on the above modifications was acceptable. The craft demonstrated a good questioning attitude during the removal of the Metal Impact Monitoring Recorder which resulted in a Field Change to the DCP. Engineering was also timely in providing an acceptable solution to the problem.

E1.2 <u>DCP S94-2-8752; Replacement of Service Water System William Powell Gate Valves (IP 37551)</u>

The inspector reviewed the DCP, UFSAR requirements for containment isolation valves, TS requirements, and the valve technical data. The inspector observed selected portions of the following:

- Removal of the old valves
- Preparation of the pipes
- Installation of the new valves
- Testing and setup of the MOV actuators
- Environmental qualification of MOV actuators

Work was generally performed IAW the DCP and plant requirements. Some minor deficiencies were noted with foreign material exclusion controls on the new valves in the laydown area and "As Low as Reasonably Achievable" (ALARA) practices in the work area. These deficiencies were promptly corrected by the licensee.

E1.3 SG Level Control and Protection System Outside Design Basis (IP 71707)

After reviewing Westinghouse issued Nuclear Safety Advisory Letter 96-004 dated October 8, 1996 the licensee concluded that under certain conditions the SG level control and protection features for Units 1 and 2 do not meet Section 4.7.3 of IEEE-279 as required by 10 CFR 50.55a(h). On November 7, SNC promptly notified the NRC, including the senior resident inspector, pursuant to 50.72(b)(1)(ii)(B) for a condition outside the design basis. Unit 1 steam flow channel selector switches were selected to the channel IV position to eliminate the problem of a common tap for the steam flow transmitter and narrow range SG water level that exist for channel III. Resident inspectors verified selector switch position and caution tags on steam flow channel switches for Unit 1. Unit 2 has the same design problem but is currently shutdown for U2RF11. The licensee has declared Unit 1 steam flow and SG water level protection features operable, and plans to resolve Unit 2 design problem with a longterm fix prior to startup. Resident inspectors will review the Unit 1 operability determination and continue to follow up on licensee longterm corrective actions for Unit 2. This issue is identified as URI 50-348, 364/96-13-04, Common Tap For SG Steam Flow Transmitter And SG Narrow Range Water Level System Fails To Meet IEEE-279.

E1.4 Control Rod Test and Evaluation Program

On October 12 through 14, 1996, the engineering support group and Operations conducted FNP-0-ETP-3661, Control Rod Test and Evaluation Program, in order to accomplish special rod control tests, evaluations and reporting requirements committed by the SNC response to NRC Bulletin 96-01, Control Rod Insertion Problems. A resident inspector observed the conduct of ETP-3661, Appendix A, Observation Of Timely Rod Insertion During Reactor Trip, as documented in NRC IR 50-348, 364/96-09. All control rods were observed to insert properly. Appendix B, Multiple Rod Testing Using Automated Measurement System Equipment, and Appendix E, Control Rod Recoil Verification, were subsequently performed while still in Mode 3, prior to cooldown for U2RF11, but not observed by the inspector. The completed test procedure and test results were reviewed by the inspector and found to be satisfactory. No abnormal control rod drop characteristics were indicated. Additional control drag testing by the vendor in the SFP and control rod drop testing using FNP-2-STP-112, Rod Drop Time Measurement, in Mode 3 was planned during U2RF11 to complete licensee NRC Bulletin 96-01 commitments.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) IFI 50-348, 364/94-28-01, Evaluation of Settings for Copes-Vulcan MOVs 8811A/B and 8812A/B Using the EPRI PPP Model.

This item was opened to track the licensee's completion of a commitment to evaluate the settings of seven 14-inch solid wedge Copes-Vulcan gate valves using criteria developed in the Electric Power Research Institute (EPRI) Performance Prediction Program (PPP). The licensee's commitment was made in response to an NRC concern regarding the reliability of the valves' settings. The settings were questioned by the NRC because they had been based on a method that required extensive extrapolation. The subject valves were employed for RHR sump suction isolation and were identified 1-8811A, 1(2)-8811B, and 1(2)-8812A/B. Their active safety function was to open.

In the current inspection, NRC inspectors reviewed and assessed the documented actions which the licensee had taken to meet their commitment. This review included Calculation SM-90-1653-018, Rev. 0, which determined opening thrust requirements for the seven valves; and a letter identified File: ENG 15 90-1653, Log: FP 96-0346, which evaluated the results of the calculation. The inspectors checked a sample of the calculation inputs and verified that they were consistent with values given in the licensee's UFSAR and design-basis document. They also verified that the licensee's calculation employed criteria developed by the EPRI PPP and that the calculation had been independently verified by a separate organization. Further, the inspectors performed an independent hand calculation which confirmed the accuracy of a "cracking" computation included in the licensee's calculation.

The licensee's evaluation included comparisons of previously determined valve capabilities with the opening thrust requirements determined in Calculation SM-90-1653-018. These comparisons showed that the present capabilities and settings of five of the seven valves were adequate, as they exceeded the opening thrust requirements. However, the comparisons showed that the other two valves did not have sufficient reduced voltage capabilities to provide the opening "cracking" (unseating) force requirements under worst case design accident conditions. The capabilities of these two valves were shown to be 0.4 and 6 percent less than required. (Note: The torque switches of the valves had been bypassed for the unseating or "cracking" portion of the opening stroke, such that the valves' full reduced voltage capabilities were not restricted by torque switch settings.) To support the adequacy of these two valves to perform their safety functions, the licensee provided a rationale that the actual cracking force requirements for the Farley valves were lower than determined in the EPRI PPP calculation. Also, they stated that the valves' reduced voltage capabilities were actually higher than initially considered, based on the measured stem friction coefficients for each of the valves. The inspectors reviewed supporting licensee data and considered it insufficient to support lower cracking force requirements. However, the inspectors found that the licensee's test results supported a dynamic stem friction coefficient of 0.18. Reduced voltage capabilities calculated using this stem friction coefficient were sufficient to provide the required cracking forces. The reduced voltage capability calculated by the licensee for the worst case valve (1-8811A) using this stem friction coefficient exceeded the calculated minimum opening thrust requirement by approximately 1 percent. The other six valves had estimated capabilities more than 10 percent greater than required (much greater than 10 percent in most cases).

The inspectors concluded that the licensee had satisfactorily completed their commitment for these valves and that this provided additional support for the capability of the valves to perform their active safety function. On this basis the IFI was closed. However, the inspectors noted continued weakness in the licensee's support for the capabilities of these valves because of the following:

- The EPRI PPP criteria used by the licensee has not yet been demonstrated to satisfactorily apply to valves manufactured by Copes-Vulcan.
- Even assuming the EPRI PPP criteria was applicable, the licensee's data for valve 1-8811A only supported a limited margin of thrust capability above that required to perform its safety function.

E8.2 (Closed) IFI 50-348, 364/94-28-02, Evaluation of Settings for Westinghouse Unit 2 MOV 8811A Using the EPRI PPP Model.

This item was opened to track the licensee's completion of a commitment to evaluate the thrust setting for a 14-inch Westinghouse flexible wedge gate valve using criteria developed in the EPRI PPP. The licensee's commitment was made in response to an NRC concern that the setting was based on EPRI PPP data, but had not been determined IAW the related criteria which EPRI had under development in the PPP. The subject valve was employed for RHR sump suction isolation and was identified 2-8811A. Its active safety function was to open.

In the current inspection, NRC inspectors reviewed and assessed the documented actions which the licensee had taken to meet their commitment. This review included Calculation SM-90-1653-019, Rev. 0, which determined opening thrust requirements for the valve; and a letter identified File: ENG 15 90-1653, Log: FP 96-0346, which evaluated the results of the calculation. The inspectors checked a sample of the calculation inputs and verified that they were consistent with values given in the licensee's UFSAR and design-basis document. They also verified that the licensee's calculation employed criteria developed by the EPRI PPP and that the calculation had been independently verified by a separate organization.

The licensee's evaluation compared the previously determined valve capability with the opening thrust requirement determined in Calculation SM-90-1653-019. This comparison showed that the present thrust setting for the valve was satisfactory, as it exceeded the opening thrust requirement determined in the calculation.

The inspectors concluded that the licensee had satisfactorily completed their commitment for this valve and that the results confirmed the adequacy of the setting used. On this basis the IFI was closed.

E8.3 (Closed) IFI 50-348, 364/94-28-03, Evaluation of Settings for Pratt Butterfly MOVs Using the EPRI PPP Model.

This item was opened to track the licensee's completion of a commitment to evaluate the settings for 16 butterfly valves manufactured by Henry Pratt. The evaluation was to be performed using criteria developed in the EPRI PPP. The licensee's commitment was made in response to an NRC concern that the licensee did not have any useful diagnostic data to support the settings used for these butterfly valves. Torque requirements for the valves had been established using guidance from the valve manufacturer. The valves were divided into groups. Their functional names and valve numbers were as follows:

Functional Name	Valve Numbers
Turbine Building Service Water Isolation Valves	1(2)-0514, 1(2)-0515, 1(2)-0516, and 1(2)-0517
Steam Generator Heat Exchanger and Boron Thermal Regenerative Chillers Service Water Isolation Valves	1(2)-3149 and 1-3150
Component Cooling Water Valves to RHR Heat Exchanger	1(2)-3185A/B

In the current inspection, NRC inspectors reviewed and assessed the documented actions which the licensee had taken to meet their commitment. This review included Calculations SM-90-1653-017, SM-90-1653-014, SM-90-1653-015, and SM-90-1653-016, which determined the torque requirements for these valves. Additionally, the inspectors reviewed the licensee's evaluations of the results of the above calculation, which were documented in a letter identified File: ENG 15 90-1653, Log: FP 96-0346.

The inspectors checked a sample of the inputs to the calculations and verified that they were consistent with values given in the licensee's UFSAR and design-basis document. They also verified that the licensee's calculation employed criteria developed by the EPRI PPP and that the calculation had been independently verified by a separate organization.

The licensee's evaluations compared the valves' torque settings with the torque requirements determined in the EPRI PPP calculations. This comparison showed that the present torque settings were satisfactory, except that the torque settings for the turbine building SW isolation valves were too low in one accident scenario. The torque switches for these valves had been bypassed in the region of concern and the valves were capable of performing their safety function. The inspectors found that the licensee had initiated documentation (e.g., WOs 450972, 450973, 450974, and 450975) to reset the torque switches to the higher values determined by the calculation.

The inspectors concluded that the licensee had satisfactorily completed their commitment for these valves and that the results showed the valves were acceptable for operation. On this basis the IFI was closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls

a. Inspection Scope (IP 83750)

The inspectors discussed planning and observed implementation of selected RWP requirements associated with the following routine tasks and U2RF11 outage job evolutions.

- RWP 096-0081, Waste Processing, Revision (Rev.) 0, effective January 1, 1996.
- RWP 296-0154, Special Plant Maintenance, All Work Associated with Primary Steam Generator Manway and Diaphragm Removal and Installation in Containment, Refueling, Rev. 0, effective October 1, 1996.
- RWP 296-0161, Refueling, Rev. 0, effective October 1, 1996.
- RWP 296-0196, Special Plant Maintenance, Work Associated with the Boron Injection Tank (BIT) Removal, Rev. 0, effective October 1, 1996.
- RWP 296-0198, Special Plant Maintenance, Work Associated with Replacement of Service Water Isolation Valves & Actuators to Containment Coolers 2A, 2B, 2C, 2D and the Reactor Coolant Pump Motor Air Coolers, Rev. 0, effective September 24, 1996.
- RWP 296-0199, Special Plant Maintenance, Work Associated with Replacement of Service Water Isolation Valves & Actuators to Containment Coolers 2A, 2B, 2C, 2D and the Reactor Coolant Pump Motor Air Coolers, Rev. 0, effective October 1, 1996.

Job planning and pre-job briefings were discussed and evaluated for selected RWP guidance. In addition, the inspectors observed licensee radiation surveys conducted adjacent to the transfer canal outside of containment and reviewed the adequacy of auxiliary building controls and postings in place during transfer of irradiated fuel.

The inspectors made frequent tours of the RCA, and reviewed and discussed specific procedural guidance, selected survey results and postings. The site dose expenditure and dose expenditure for selected U2RF11 tasks were reviewed and discussed with HP supervisors and technicians.

In addition, FNP OR Number 96-227 dated October 7, 1996, documenting details of a worker who exceeded plant administrative quarterly dose limits and an associated discrepancy between his thermoluminescent dosimeter (TLD) and digital alarming dosimeter (DAD) monitoring results were reviewed and discussed. The status of licensee followup actions regarding the event were discussed in detail.

b. Observations and Findings

Excluding two identified events involving BIT system maintenance and laundry processing, all work activities observed were conducted IAW the established RWP requirements. Initial reviews and surveys to establish controls and postings were IAW FNP-O-RCP-4, Refueling Survey, Rev. 13a, dated October 21, 1996. Administrative and physical controls, and established postings within the auxiliary building during movement of irradiated fuel were verified to be adequate based on measured dose rates.

TS 6.11 requires, in part, that procedures for personnel radiation protection be prepared consistent with the requirements of 10 CFR Part 20 and be approved, maintained and adhered to for all operations involving personnel radiation exposure. Procedure FNP-O-M-001, Health Physics Manual, Rev. 12, effective July 14, 1996, Section (§) 6.4 requires any entry into the RCA to be governed by a RWP. During tours of the RCA during the week of October 21, 1996, the inspectors identified the following issues.

- On October 22, 1996, a worker performing maintenance on the BIT recirculation pump equipment located adjacent to the auxiliary building 100 foot elevation batching area, was observed kneeling and crawling within an area posted as "contaminated" without the required coverall dressout specified by RWP-096-0196, Special Plant Maintenance, effective October 1, 1996.

- On October 24, 1996, a HP support worker was observed operating the automated laundry monitor (ALM) without proper gloves and shoe covers as required by RWP-096-0081, Waste Processing, Rev. 0, effective January 1, 1996.

Initial corrective actions for the identified RWP compliance concerns included work stoppage, counseling of individuals involved and subsequent discussions with supervisors. None of the involved individuals were found to be contaminated during RCA exit surveys. For the BIT maintenance work issue, the inspectors requested the licensee to conduct contamination surveys within the posted contaminated area where the work was observed on the auxiliary building 100 foot elevation. Licensee survey results indicated contamination levels (beta/gamma) were less that 200 disintegrations per minute per 100 square centimeters. Discussions with HP staff indicated that the area was posted as a contaminated area due to the potential for release of contamination from within the BIT equipment during the maintenance activities. The licensee initiated FNP OR Nos. 96-1002 and 96-1003 for the identified RWP compliance concerns. On October 25, 1996, licensee representatives informed the inspectors that preliminary review indicated that observed improper dressout observed for the ALM operations resulted from misinterpretation of RWP requirements by the responsible supervisor of the involved worker.

On October 25, 1996, the inspectors attended and observed a pre-job briefing associated with removal of the SG manway diaphragms. Meaningful discussions between HP and maintenance staff were noted regarding changes to previous storage practices following diaphragm removal. Previous use of large shielded containers were discontinued with new storage to be provided using drums containing water for shielding. The use of the water filled drums improved dose reduction efforts and reduced safety concerns associated with movement of the large containers within containment. From subsequent discussions with licensee representatives regarding this change, the inspectors noted that although evaluated, the licensee had evaluated the change in storage methods qualitatively, a detailed evaluation of the dose reduction effect using the new water filled drums was not documented.

The inspectors reviewed and evaluated licensee actions associated with FNP OR No. 96-27, dated October 7, 1996. The report documented an HP support individual's dose as measured by TLD of 1122 millirem (mrem) which exceeded the established administrative quarterly limit of 1000 mrem. Licensee followup also identified a significant difference, approximately 44 percent, between the TLD and the DADs used to monitor the worker's dose for the quarter. On average, TLD to DAD monitoring result comparisons were less than five percent. The worker was excluded from further RCA entries and an investigation initiated. Review of the individual's daily DAD entries, indicated maximum potential for exposure to elevated dose rates occurred on September 6 and 10, 1996, during placement of spent filters in High Integrity Containers in a Radwaste exclusion area. General license followup included interviews of all personnel involved and review of radiation control practices associated with the subject filter placement; verification of DAD calibrations and calibrator operation, and confirmation of TLD readings; and determination and evaluation of filter isotopic data and energy response of personnel monitoring equipment. The licensee had assigned the subject individual the 1126 mrem exposure resulting in a year-to-date (YTD) total effective dose equivalent (TEDE) exposure of 1160 mrem. At the end of the onsite inspection, no definite causes for the observed differences between TLD and DAD quarterly dose results were identified and licensee evaluations were continuing.

As of October 24, 1996, maximum TEDE results, 1160 and 806 mrem, were reported for two individuals involved with handling spent filters within the radwaste facilities in September 1996. Extremity shallow dose exposure results of 5762 and 5414 mrem were reported for these individuals. For the outage activities reviewed, the maximum individual dose, i.e., deep dose equivalent, of 300 mrem was documented for a contractor involved with eddy current testing.

c. Conclusions

In general, RWP guidance was adequate for routine RCA and U2RF11 outage activities. Documentation of detailed dose reduction efforts and calculations should be improved. Excluding one individual, all personal exposures were less than administrative limits and all individuals were within regulatory limits. Licensee review and followup actions for a worker exceeding quarterly administrative dose limits were adequate. Two examples of inadequate implementation of RWP dressout requirements were observed. These examples were identified as Violation (VIO) 50-348,364/96-13-05, Failure to Follow Radiation Work Permit For Use of Proper Protective Clothing .

R1.2 Implementation of the ALARA Program

a. Inspection Scope (IP 83750 and 84750)

The licensee's ALARA program guidance and implementation associated with the current outage activities were discussed and reviewed. In addition, selected radiation control performance indicators were reviewed and discussed with licensee representatives.

b. Observations and Findings

For 1996 which included a single refueling outage, licensee representatives established a dose goal of 250 rem. This dose expenditure for the site is similar to the 251 rem expended in 1994, also single outage year. As of October 23, 1996, the licensee YTD dose expenditure was documented as 10.503 rem compared to a predicted exposure of 43.925 rem. The difference was expected, in part, as a result of delays in the outage schedule.

The amount of contaminated floor space, excluding containment and exclusion areas, was reported as less than or equal to approximately eight percent since 1994. As of September 27, 1996, licensee listed 5.5

percent of floor space as contaminated, a slight decrease from 5.7 percent reported in Jan 26, 1996. Similar values, 5.26 and 8.38 percent were reported for April and July 1995, respectively. For 1994, the licensee reported contaminated floor space ranging from 6.0 to 8.2 percent.

For 1996, only one personnel contamination event (PCE), defined as contamination levels exceeding 5000 disintegrations per minute, was reported. For single and dual outage years of 1994 and 1995, the licensee listed 55 and 74 PCEs, respectively. The majority of PCEs were identified for contractors and were associated with poor work practices.

The inspector reviewed ALARA initiatives conducted during the U2RF11 outage IAW the long-term exposure reduction program outlined in FNP Exposure Reduction Plan, dated May 1993. The inspector discussed and verified implementation of the following ALARA program items including cobalt reduction, crud trap flushing, early boration, elevated pH and boron/lithium management, improved ALARA training and awareness, remote personnel monitoring, robotics and selected SG maintenance enhancements. The licensee had suspended the zinc injection program since the U2RF9 outage but was expected to resume its implementation during the current outage.

c. Conclusions

Implementation of established ALARA program activities was verified. No significant negative trends were observed for the performance indicators reviewed.

R2 Status of (RP&C) Facilities and Equipment

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

During the course of the inspection period the resident inspectors conducted numerous tours of the auxiliary building RCA for Units 1 and 2. In general, HP control over the RCA, and the work activities conducted within it, were good. Material condition and housekeeping in the Unit 1 and 2 RCA, considering ongoing outage activities, were much better maintained than in the past (see Section O2.1).

R2.2 General walkdowns of radiation monitoring systems

a. Inspection Scope (IP 83750 and 84750)

The inspectors reviewed and evaluated general housekeeping and verified, where applicable, operability of selected process and effluent RMS detectors, electronics, sampling lines and flow meters. The following RMS samplers or detectors, i.e. radiation elements (REs), and associated equipment were included in the walkdowns: Unit 1 (U1) containment atmosphere particulate (RE-11) and gas (RE-12); U1 turbine building ventilation exhaust normal range (RE-15); U1 & U2 plant vent gas (R-29B) and particulate (RE-29A); MCR air supply (RE-35A&B); and U1 & U2 post accident sampling system airborne particulate (RE-67).

In addition, the inspectors reviewed and discussed program guidance and testing of air to ensure service air compressor system supplied Grade D respirable air for use, as applicable, during U2RF11 job evolutions.

b. Observations and Findings

In general, housekeeping practices associated with RMS detectors and equipment were improved relative to conditions observed for the period of August 12-16, and 26-30, 1996 and documented in IR 50-348, 364/96-10, dated September 27, 1996. Only two examples of poor housekeeping practices associated with the RMS equipment skids or sample locations were noted during equipment walkdowns conducted on October 25, 1996. The examples included unsecured equipment stored within the U1 plant vent particulate (RE-29A) sampler area and excess filter papers being stored within the U1 Plant Vent sampler cabinet (RE-29B). Licensee representatives stated that the identified concerns would be corrected in a timely manner.

The inspectors also verified that the service air compressor system was tested to certify supplied breathing-air as Grade D for potential use during outage activities. Licensee representatives collected Unit 2 containment breathing system air samples on October 15, 1996, IAW FNP 2-RCP-112, Sampling of Service Air to Meet Respiratory Limits, dated September 9, 1996. Sample results verified that the U2 containment air quality exceeded the established limits for Grade D air based on the Compressed Gas Commodity Specification G7.1, 1973.

c. Conclusions

Overall, no significant concerns were identified for RMS operability and for certification of the supplied breathing air equipment. Licensee tests verified that the service air compressor system supplied Grade D respirable air IAW 10 CFR 20, Appendix A requirements.

R3 RP&C Procedures and Documentation

a. Inspection Scope (IP 83750)

Records of the previous 1996 YTD occupational radiation doses for approximately 20 contractors hired specifically for U2RF11 outage tasks were reviewed and discussed with licensee staff.

The inspectors also reviewed selected effluent release data. In particular, the inspectors reviewed and discussed abnormal effluent releases documented for the period 1994 through October 21, 1996.

b. Observations and Findings

The inspectors verified that all contractor personnel had provided a signed up-to-date NRC Form 4, or equivalent prior to conducting initial work within the RCA. Further, the licensee had received, or was in the process of obtaining reports of each individual's previous 1996 dose equivalents from the most recent employers for work involving radiation exposure.

For 1994 no abnormal effluent releases were identified. In 1995 only one abnormal effluent release associated a small U1 primary to secondary leak and the venting of the atmospheric relief valves following a U1 reactor trip was identified and evaluated. The event as documented in chemistry incident report (CIR) 1-95-009, resulted in a increase in offsite dose. For 1995, dose estimates from all effluents were a fraction of a percent of the allowable offsite dose calculation manual (ODCM) limits. As of October 21, 1996, three abnormal effluent releases were reported and documented in CIRs 1-96-19, 1-96-30, and 1-96-33. The inspector noted that CIR 1-96-19 documented and calculated a negligible dose effect from required surveillance testing of the U1 TDAFW pump during a small primary to secondary leak. The two remaining CIRs identified a continuous leak of effluents containing tritium through the U1 MS line atmospheric relief valve. Effects of the leak were to be documented in the 1996 annual effluent report.

c. Conclusions

Licensee records of previous dose estimates for outage contractors were being completed IAW 10 CFR 20.2104 requirements. Documentation regarding effluent releases was prepared IAW ODCM requirements and demonstrated release data verified offsite doses were small fraction of a percent of the allowed limits. The number of abnormal releases have increased since 1994, mainly, as a result of U1 primary to secondary leakage associated with required or inadvertent equipment operation, or maintenance problems. These abnormal releases had a negligible contribution to offsite doses.

R6 RP&C Organization and Administration

a. Inspection Scope (IP 83750)

Recent changes to the HP organization and staffing levels for the U2RF11 outage relative to the 1995 U2RF10 were reviewed and discussed. From facility tours and observations of work in progress the inspectors evaluated staffing adequacy and proficiency of the HP technicians providing job coverage.

b. Observations and Findings

The HP organization and permanent staff for routine operations have remained relatively stable. Since a previous U2 refueling outage in 1995, the only organizational change involved combining the plant health physicist and radwaste supervisor positions and elimination of a permanent HP technician position. Currently, the permanent HP staff included a HP superintendent, a HP supervisor, health physicists/radwaste supervisor, five HP foremen and 33 senior technicians. During the current outage, the licensee supplemented the staff with approximately 50 contract HP personnel and 21 HP and chemistry staff members from Vogtle and Hatch Nuclear Plants. Overall, the numbers of non-permanent HP personnel employed during the current outage decreased by approximately 15 technicians, approximately ten junior and five (senior) HP technicians since the previous U2RF10 outage.

No concerns regarding job coverage nor HP technician proficiency were identified during observation of fuel movement, RHR pump maintenance, and SW valve replacement activities.

c. Conclusions

No concerns were identified for the current organization. The amount of job coverage and proficiency of HP technicians were considered adequate for the early outage tasks observed.

R7 Quality Assurance in RP&C Activities

R7.1 Licensee Self-Assessment Activities

a. Inspection Scope (IP 84750)

The inspectors reviewed and discussed the detailed plans for audits scheduled to be conducted during the U2RF11 outage. The following audit plans were reviewed in detail.

- Safety Audit and Engineering Review (SAER) Audit Report No. 96-OA/41-1, Refueling Outage Activities
- SAER Report No. 96-PRM/23-1, Primary Vendor Services Audit
- SAER Report No. 96-FL/27, Fuel Loading
- SAER Report 96-PRTP/32, Post Refueling Test Program

In addition, supplemental audit staffing was reviewed and discussed.

b. Observations and Findings

Review of the audit plan details verified that radiological protection issues were included in the SAER evaluations scheduled for the current outage. Audit plan details included, in part, observation and verification of radiological work controls for routine outage activities and radiography practices, internal exposure controls, housekeeping and cleanliness, personal qualifications and certifications, and verification of completion of Standard Test Procedures conducted during the outage. From review of audit plans and discussions with SAER auditors, the inspectors determined that specific job evolutions to be reviewed included radiological controls associated with U2 fuel movement, SG maintenance and SW valve replacement activities. The inspectors directly observed auditors conducting radiological work practice observations for auxiliary building SW valve replacement and U2 containment fuel movement activities.

An additional outside auditor with senior reactor operator (SRO) experience was scheduled to assist the FNP SAER group during the current outage. In addition, SAER management stated that in response to concerns addressed in NRC IR 50-348/96-10, 364/96-10 dated September 27, 1996, an individual from the Vogtle Nuclear Plant with extensive chemistry and radiation protection experience was scheduled to participate in a future RP&C audit in early 1997.

c. Conclusions

No concerns were noted for ongoing and proposed audits of radiation control and chemistry activities. The scheduling of outside auditors to assist in review and evaluation of RP&C program areas was considered a program enhancement.

R8 Miscellaneous RP&C Controls Issues (92904)

R8.1 (Closed) LER 50-348, 364/95-06: Licensed Material Shipped to Incorrect Destination by Common Carrier

This LER was a minor issue and was closed.

S1 Conduct of Security and Safeguards Activities

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

During routine inspection activities, resident inspectors verified that portions of site security program plans were being properly implemented. This was evidenced by: proper display of picture badges by plant personnel; appropriate key carding of vital area doors; adequate stationing/tours of security personnel; proper searching of packages/personnel at the primary access point and service water intake structure; and adequacy of compensatory measures (i.e., posting of guards) during disablement of vital area barriers. Security activities observed during the inspection period were well performed and appeared adequate to ensure physical protection of the plant. Guards were observed to be alert and attentive while stationed at disabled doors and access covers to critical underground equipment (e.g., SW system valve boxes). Posted positions were manned with frequent relief.

S8 Miscellaneous Security and Safeguards Issues (IP 71750)

S8.1 (Closed) URI 50-348, 364/96-09-05: Failure to Search Contractor Trailer Prior to Entry Into the Protected Area (PA)

On October 10, 1996, a resident inspector observed security guards escort a Westinghouse sludge lance trailer into the PA that was not searched. The trailer was posted as a RCA. A security guard outside the PA gate did search the truck, cab, and driver prior to entering the PA. After subsequent review of licensee corrective actions, and interviews with responsible individuals and supervision, the inspector concluded that this instance constituted a violation of the FNP Security Plan, section 4.4.2 that requires searching all vehicles, materials and packages prior to entering the PA, with certain exceptions established as Categories I - IV. Categories I and III would allow certain types of materials or packages to enter the PA without being searched as long as they were under continuous direct observation, or positive controls were put in place, respectively. Categories II and IV did not apply to this situation. Also it was the inspectors judgement that the RCA boundary around the trailer did not constitute a personnel hazard per Category II. The other categories did not apply to this situation.

Security guards did not search the Westinghouse sludge lance trailer prior to entering the PA, and subsequently relinquished direct observation of the trailer without establishing positive control. A search was not conducted until the following day on October 11. Upon notification of the problem, immediate corrective actions were taken by the Security Chief to promptly and effectively address the problem. By the end of the IR period, longterm corrective actions were still being pursued that should considerably improve effectiveness of future PA searches. Good coordination was evident by the Security Chief with other FNP departments and outside sources in developing a new permanent policy. This issue is identified as VIO 50-348, 364/96-13-06, Failure To Search Truck Trailer Prior To Entering Protected Area, and closes this URI.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Operability of Fire Protection Facilities and Equipment

a. Inspection Scope (IP 64704)

The inspectors reviewed the open maintenance WOs, maintenance history, and incident reports on the facilities fire protection systems and features, and inspected these items to determine the performance trends and the material conditions of this equipment.

b. Observations and Findings

Maintenance Observations:

As of November 12, 1996, the total number of open maintenance work requests related to the fire protection systems and features was 81. These work requests were grouped as follows:

Kaowool Fire Barriers	38
Fire Protection Water Systems	34
CO_2	5
Fire Doors	2
Fire Pumps	1
Fire Detection System	1
	81

All except five of these work requests were issued in 1996. The work requests issued prior to 1996 were minor repairs which did not affect the operability of these systems. The Kaowool work requests involved a number of recently identified discrepancies. Work was in process to correct these issues.

There was not a backlog of open work requests.

Fire Protection Related Incident Reports:

The licensee initiated 54 incident reports from January 1, 1993 through October 31, 1996, on fire protection related items, such as fire pumps, automatic sprinkler systems, fire detection system, CO_2 systems, fire barriers, and fire watch activities. These incident reports were as follows:

Fire Protection System/Feature	Number	Percent
Fire Pumps	14	25.8
Fire Watch	9	16.4
Sprinkler and Fire Hose Systems	9	16.4
Fire Alarm System	8	14.6
CO_2 Systems	8	14.6

Personnel Errors	4	7.1
Fire Barriers	2	3.5
Fire Doors	1	1.6
Totals	55	100.0

Incident reports related to the recent Kaowool problems where not included in this list.

Most of the abnormal occurrences, except for the Kaowool problems, during this period were related to problems with the fire pumps, fire watches and water suppression systems.

Fire Protection System Operability:

A review of the Fire Protection portion of the Plant's Daily Status Report for November 12, 1996 indicated the following components or systems were out of service:

Fire Protection System	Unit 1	Unit 2
Kaowool Fire Barriers	49	24
Fire Doors	8	16
Fire Barrier Penetrations	3	6
Automatic Sprinkler Systems	1	12
Fire Hose Stations	0	1
Fire Detection System	1	2

The inspector considered the number of fire protection systems out of service to be excessive. However, this high number was attributed to the current Unit 2 refueling outage and the repairs in process for the Kaowool fire barrier discrepancies. Appropriate compensatory measures had been implemented for the equipment which was out of service.

The status report provided the licensee with a good means of identifying out of service fire protection equipment to assure that appropriate compensatory measures were implemented.

During the plant tours, the inspector noted that the maintenance and material condition of the fire protection equipment were satisfactory, except for a significant number of pre-action automatic sprinkler valves which were set wet. These valves are designed to be maintained in the close position and activate to the open position by signal from the associated fire or smoke detection system. When these valves are maintained in the tripped or open position, the water flow alarms to these systems are placed out of service and sprinkler system actuation would not be transmitted by the alarm system. This increases the possibility of water damage to plant equipment in the event of inadvertent actuation of these systems. Leaving the pre-action valves in the tripped position for appreciable long periods of time has been a normal practice at Farley for several years. This is identified as a program weakness.

In March 1996, as documented by NRC IR Nos. 50-348, 364/96-02 and 96-07, multiple failures occurred during the routine operability testing of the pre-action automatic sprinkler system valves. Approximately 9 of the 27 pre-action system valves installed to provide fire protection for safety related areas failed to operate automatically upon an actuation signal from the fire detection system, i.e. the valves would not operate from the normally closed to the open water flow position. The licensee implemented additional preventive maintenance measures for these valves, contacted the vendor for assistance and scheduled an accelerated surveillance testing program. The surveillance testing of these valves was changed from 18 months to two months. Prior to the August 1996 scheduled tests, two valves failed to operate on August 12 following an inadvertent action of the fire alarm panel. On August 22 during performance of the two month accelerated surveillance testing activities, one additional valve failed to operate either manually or automatically. The licensee assembled a root cause team and continued to work with the vendor to determine the cause of these failures. One of the failed valves was sent to the vendor's test laboratory for further evaluation. The results of this evaluation were not available at the conclusion of this inspection. The vendor was scheduled to participate during the next surveillance testing of these valves in December 1996. The reliability of these valves is considered questionable until the licensee identifies the cause of these failures and implements appropriate corrective action to resolve the problem. This issue is being tracked as IFI 50-348, 364/96-02-03, Pre-action sprinkler system failures.

From January 1993 through October 1996, multiple failures of the diesel engine driven fire pumps to start on demand were identified or diesel engines had to be shut down due to operability problems. Examples of these problems included: inoperable electrical selector switches, electrical starter switch, engine starter, engine batteries, leaking engine coolant hose connector, ruptured coolant hose connector, and leaking oil from the lubrication system. The inspector reviewed approximately 14 incident reports which had been issued on the diesel driven fire pumps.

To establish a high confidence level on the operability of these pumps, the automatic start surveillance test for the diesel driven pumps was changed in May 1995 from monthly to every two weeks and the frequency of the functional and capacity tests was changed from 18 months to annually. The last recorded failure of a diesel driven fire pump was September 26, 1995. The licensee provided the inspector with trending information which indicated that the performance of the two diesel driven fire pumps in the 18 months prior to November 1996 had improved as follows:

Diesel Driven Fire Pump 1	Three start failures in previous 228 demands for a reliability of 98.7%
Diesel Driven Fire Pump 2	No start failures in previous 195 demands for a reliability of 100%

The licensee's program to establish a high level of confidence in the operability of these pumps was considered pro-active.

The licensee informed the inspector that replacement parts for the station fire alarm control panels were no longer manufactured and were becoming difficult to obtain. The existing system was operable but due to the lack of replacement parts, future reliability may be a problem.

c. Conclusions

The number of outstanding work requests related to the fire protection systems was high. However, there was not a backlog of outstanding WOs. Corrective maintenance on degraded fire protection systems was being accomplished in a timely manner. The fire pumps and automatic sprinkler systems sustained reliability problems during the past two years due to a number of operational failures. The licensee had taken positive corrective action initiatives to resolve these concerns. This action was effective for the fire pumps but currently has not been effective on the resolution of the problems with the automatic sprinkler systems. Leaving the pre-action valves in the tripped position for long periods of time has been a normal practice at Farley for several years. This is identified as a program weakness.

Replacement parts for the site fire alarm system were difficult to locate. The system was operable but reliability may decline due to lack of readily available replacement parts.

The daily fire protection status report was considered a positive means of identifying degraded fire protection systems and to implement the appropriate compensatory measures for inoperable systems.

F2.2 Surveillance of Fire Protection Features and Equipment

a. Inspection Scope (IP 64704)

The inspectors reviewed the surveillances and tests scheduled for the various fire protection systems and features to determine compliance with UFSAR Section 9B Attachment C.

b. Observations and Findings

Available documentation or cross reference material was not available to indicate that all of the tests and inspections listed by UFSAR Section 9B Attachment C had been incorporated into appropriate plant surveillance procedures. Therefore, the inspector selected 18 fire protection inspection and surveillance requirements from the UFSAR to verify that these items had been incorporated into the surveillance procedures. It was noted that the operability test of the automatic fire and smoke detectors did not meet the frequency listed by the UFSAR.

UFSAR Section 9B.C.1.1.2 requires accessible smoke detectors to be demonstrated operable once per six months. The licensee had recently changed this test requirement. The new test requirement was to demonstrate fire and smoke detector operability once per two years.

The licensee provided two 10 CFR 50.59 Evaluation Reports, Diesel Building Fire Detector Surveillance Frequency Revision dated September 24, 1996, and Change for Frequency of Smoke Detector Testing dated November 21, 1995. These evaluations used past satisfactory test results and the bench test, sensitivity calibration and enhanced cleaning and maintenance program performed on each detector every two years as justification for changing the test frequency. In addition, the evaluation indicated that trending of the detector test program and failures were to be monitored. If the failure rate increased, the two year test frequency would be adjusted accordingly to ensure that adequate reliability of the fire detection system was maintained. Therefore, the justification provided to change the testing frequency from six months to two years was appropriate.

c. Conclusion

The surveillance and tests of the fire protection systems and features met the frequencies specified by UFSAR Section 9B Appendix C, except for the functional operability test of the fire and smoke detector instrumentation. The frequency of these tests was recently changed from six months to two years. The evaluation performed by the licensee to justify this change was appropriate.

F3 Fire Protection Procedures and Documentation

a. Inspection Scope (IP 64704)

The following Station Administration Procedure and Fire Protection Procedures were reviewed for compliance with the NRC requirements and guidelines:

- FNP-0-AOP-29.0, Revision 13, Plant Fires
- FNP-0-EIP-13, Revision 14, Fire Emergencies
- FNP-0-EIP-3401, Revision 3, Transient Fire Load Analysis
- FNP-0-AP-35, Revision 20, General Housekeeping and Cleanliness Control

- FNP-0-AP-36, Revision 12, Fire Surveillance Procedures and Inspections

- FNP-0-AP-37, Revision 11, Fire Brigade Organization

- FNP-0-AP-38, Revision 10, Use of Open Flame

- FNP-0-AP-39, Revision 12, Fire Patrols and Watches

- FNP-0-AP-45, Revision 15, Training Plan
  Appendix P, Fire Brigade Training Program
  Appendix Q, Fire Brigade Retraining Program

- FNP-0-AP-63, Revision 5, Conduct of Operations, Engineering Support Department, Section 2.1.4, Fire Protection Program

Plant tours were performed to assess procedure compliance.

b. Observations and Findings

The above procedures established the administrative guidance used to implement the fire protection program at Farley and included the requirements for the control of combustibles, ignition sources and fire brigade organization and training. The procedures met the intent of the NRC requirements.

The operability, surveillance and test requirements for the fire protection systems and features had been removed from the TSs and incorporated into UFSAR Section 9B Attachment C. These requirements met the requirements for the fire protection features which were formerly in the TSs, except for the testing of the fire detection system as discussed in Section F2.2. However, an appropriate evaluation had been provided to justify this change.

The inspector performed plant tours and noted that the general housekeeping related to the control of combustibles within the plant and implementation of the other fire prevention procedure requirements were satisfactory.

c. Conclusions

The fire protection program implementing procedure met the intent of the NRC guidelines and requirements. Implementation of the fire protection and prevention procedures and the general housekeeping for control of combustibles within the plant were satisfactory.

F5 Fire Protection Staff Training and Qualification

a. Inspection Scope (IP 64704)

The inspector reviewed the fire brigade organization and training for compliance with the facility's fire protection program and the NRC guidelines and requirements and witnessed a fire brigade drill.

b. Observations and Findings

The organization and training requirements for the Farley plant fire brigade were established by Procedure FNP-0-AP-37, Fire Brigade Organization, Revision 11. The fire brigade for each operational shift was composed of a fire brigade leader (operations shift foreman) and three brigade members (non-licensed system operators) from operations and one brigade member from security. The fire brigade leader was normally a licensed SRO. Each fire brigade member was required to receive initial, quarterly and annual fire fighting related training and satisfactorily complete an annual medical evaluation to certify participation in the fire brigade. There were a total of 104 operations personnel and 24 security personnel on the plant's fire brigade.

The inspector reviewed the Training Department's training summary records and verified that the training for the fire brigade personnel was up to date. A minimum of six drills were performed each quarter and scheduled such that each fire brigade member attended at least two drills per year. Most of the fire brigade drills had been unannounced drills.

On November 13, the inspector witnessed a fire brigade drill involving a simulated fire at the fire pumps' diesel fuel tank and fire pump house. The fire brigade leader and four fire brigade members responded in full fire fighting turnout gear. Personnel from HP and operations also responded to the drill. The action by the brigade met the established drill objectives, except for some minor problems encountered with self contained breathing apparatus. However, additional equipment was available if this equipment had actually been needed. A drill critique was conducted with the fire brigade members following the drill.

c. Conclusions

The fire brigade organization and training met the facility's procedure requirements and the performance by the fire brigade to a drill during this inspection was good.

F6 Fire Protection Organization and Administration

a. Inspection Scope (IP 64704)

The licensee's management and administration of the facility's fire protection program were reviewed for compliance with the commitments to the NRC and to current NRC guidelines

b. Observations and Findings

The Plant Operations Assistant General Manager was designated as the onsite manager responsible for the administration and implementation of the fire protection program. The daily control of the fire protection program was assigned to the station Fire Marshal who reported to the Engineering Support Supervisor under the management of the Engineering Support Manager and the Plant Support Assistant General Manager.

Most of the surveillance inspections and tests, corrective and preventive maintenance for the fire protection systems and features were provided by a designated maintenance team composed of three mechanical, three electrical, and one I&C maintenance craft personnel, and two system operators (non-licensed operators). This maintenance team worked primarily on fire protection systems and other plant support systems such as heating and ventilation components and was under the supervision of the Maintenance Manager. The Fire Marshal reviewed all completed surveillance and test procedures and coordinated the maintenance work activities to assure that appropriate inspections, tests and maintenance were performed. Engineering technical support was provided as needed from the engineering personnel on site and from the corporate office staff in Birmingham, Alabama.

The responsibility for the fire brigade training was assigned to a fire brigade training instructor in the Training Department.

There did not appear to be a formal program for trending fire protection condition reports and performance of fire protection system testing. However, periodic informal interface between the Fire Marshal and various personnel assigned fire protection related functions was being made to coordinate the implementation of the fire protection program.

c. Conclusions

The coordination and oversight of the facility's fire protection program met the licensee's commitments to the NRC in the UFSAR. The personnel assigned various fire protection related functions were working together as a team and with coordination by the Fire Marshal to implement the fire protection program at the site.

F7 Quality Assurance in Fire Protection Activities

a. Inspection Scope (IP 64704)

The following quality assurance (QA) audit reports were reviewed:

Audit dated 8/24/89	Station Fire Protection Annual/Triennial Audit
Audit dated 5/30/90	Station Fire Protection Annual Audit
Audit dated 8/3/92	Station Fire Protection Annual/Biannual/Triennial Audit
Audit dated 3/30/93	Station Fire Protection Annual/Triennial Audit
Audit dated 5/17/94	Station Fire Protection Biannual Audit
Audit dated 8/19/94	Station Fire Protection Annual Audit
Audit dated 8/10/95	Station Fire Protection Annual/Biannual/Triennial Audit
Audit dated 7/22/96	Station Fire Protection Annual Audit

b. Observations and Findings

These audits were thorough and identified a number of findings, recommendations and comments for program enhancements. A different independent fire protection specialist was provided for each of the audits. This provided different perspectives of the fire protection program. The corrective actions taken on each of the audit findings, recommended enhancements and comments from each QA report were reviewed by the inspector. The corrective actions for major discrepancies or findings were found to have been completed in a timely manner. However, action on the recommendations and comments which were made to enhance the program were not addressed in a timely manner. Three of five enhancement items in the 1993 audit, three of five in the 1995 audit and two of five from the 1996 audits had not been completed. A formal program was only provided to track the completion of the corrective actions for major discrepancies.

c. Conclusions

The audits and assessments of the facility's fire protection program were thorough and corrective actions were taken in a timely manner to resolve major identified discrepancies. However, resolution on recommendations and comments to enhance the fire protection program was not timely.

F8 Miscellaneous Fire Protection Issues (IP 92904)

F8.1 Fire Protection Related NRC INs

a. Inspection Scope

The inspector reviewed the licensee's evaluation for the following NRC INs:

- IN 92-18, Potential Loss of Shutdown Capacity During a Control Room Fire
- IN 92-28, Inadequate Fire Suppression System Testing
- IN 93-41, One Hour Fire Endurance Tests Results for Thermal Ceramics, 3M Company FS-195 and 3M Company Interam e-50 Fire Barrier Systems
- IN 94-28, Potential Problems with Fire Barrier Penetration Seals
- IN 94-31, Potential Failure of WILCO, LEXAN-Type HN-4-L, Fire Hose Nozzles
- IN 94-58, Reactor Coolant Pump Lube Oil Fire
- IN 95-36, Emergency Lighting

b. Observations and Findings

The evaluations for these INs were appropriate and the appropriate actions had been completed, except for IN 93-41 and IN 95-36.

IN 93-41, One Hour Fire Endurance Tests Results for Fire Barrier Systems

A review of the licensee's evaluation of the Kaowool one hour fire barriers installed at Farley found that these barriers did not meet the NRC guidelines of Generic Letter 86-10, Supplement 1. The principle deviations from the NRC guidelines were: Kaowool was not tested by an independent laboratory in an approved large scale furnace, temperature measured on the external raceway exceeded 165° C [tested raceways were approximately 426° C], cable was damaged by the fire tests, tested configurations did not match in plant installations, and fire barriers were not subjected to a hose stream test after the fire test. Therefore, the adequacy of the installed fire barriers at Farley is being reevaluated by the NRC. This issue was previously identified as URI 50-348, 364/96-09-08 and remains open pending completion of the NRC's reevaluation.

IN 95-36, Emergency Lighting

The licensee was reevaluating this IN due to recent problems with the Appendix R, 8-hour emergency lighting units. The licensee's reevaluation of this IN will be reviewed during a subsequent NRC inspection.

c. Conclusions

The evaluations and the actions taken on the reviewed INs were appropriate, except for IN 93-41. An URI was identified for IN 93-41 concerning the adequacy of the one hour Kaowool fire barriers. The licensee was reevaluating IN 95-36 for applicability at Farley.

V. Management Meetings and Other Areas

X1 Review of UFSAR Commitments

A recent discovery of a licensee operating their 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 inspector reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with observed plant practices, procedures and/or parameters. Only one exception was identified, as follows:

- UFSAR Appendix 3K, High Energy Line Break (Outside Containment), provides no description of the pressure differential switch high detectors identified in TS 3.3.3.7 for High Energy Line Break (HELB) Isolation Sensors. Inspection followup of the licensee resolution of this omission is identified as IFI 50-348, 364/96-13-07, Certain HELB Isolation Sensors Not Described In UFSAR.

X2 Exit Meeting Summary

The resident inspectors presented the inspection results to members of licensee management on November 27, 1996, after the end of the inspection period. The licensee acknowledged the findings presented.

The resident 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

W. Bayne, Chemistry/Environmental Superintendent
R. Coleman, Maintenance Manager
S. Fulmer, Technical Manager
H. Garland, Assistant Maintenance Manager
D. Grissette, Operations Manager
R. Hill, General Manager - Farley Nuclear Plant
R. Martin, Superintendent Operations Support
M. Mitchell, Health Physics Superintendent
R. Monk, Engineering Support Supervisor - Equipment Evaluation
C. Nesbit, Assistant General Manager - Support
J. Odom, Superintendent Unit 1 Operations
J. Powell, Superintendent Unit 2 Operations
L. Stinson, Assistant General Manager - Plant Operations
J. Thomas, Engineering Support Manager
B. Yance, Plant Modifications and Maintenance Support Manager
W. Warren, Engineering Support Supervisor - Performance Review
G. Waymire, Safety Audit and Engineering Review Site Supervisor

NRC

J. Zimmerman, Project Manager - Farley Nuclear Plant

INSPECTION PROCEDURES USED

IP 37550: Engineering
IP 37551: Onsite Engineering
IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 60710: Refueling Activities
IP 61726: Surveillance Observations
IP 62703: Maintenance Observations
IP 62707: Maintenance Observations
IP 64704: Fire Protection/Prevention Program
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
IP 92904: Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
URI	50-364/96-13-01	Open	PRF Operability Requirements for SFP (Section O2.6).
IFI	50-364/96-13-02	Open	Increased Frequency Test Program for Charging Pumps due to Cladding Cracking (Section M1.5).
IFI	50-348, 364/96-13-03	Open	Foreign Material From Seal Injection System To RCP Seals (Section M1.10).
URI	50-348, 364/96-13-04	Open	Common Tap For SG Steam Flow Transmitter And SG Narrow Range Water Level System Fails To Meet IEEE-279 (Section E1.3).
VIO	50-348, 364/96-13-05	Open	Failure to Follow Radiation Work Permit For Use of Proper Protective Clothing (Section R1.1).
VIO	50-348, 364/96-13-06	Open	Failure To Search Truck Trailer Prior To Entering Protected Area (Section S8.1).
IFI	50-348, 364/96-13-07	Open	Certain HELB Isolation Sensors Not Described In UFSAR (Section X1).

Closed

Type	Item Number	Status	Description and Reference
LER	50-348/95-10	Closed	Actuation of Engineered Safety Feature Equipment Due to Loss of Main Feedwater (Section O8.1).
LER	50-364/95-08	Closed	Reactor Trip During DEH Card Changeout (Section M8.1).
IFI	50-348, 364/94-28-01	Closed	Evaluation of Settings for Copes-Vulcan MOVs 8811A/B and 8812A/B Using the EPRI PPP Model (Section E8.1).

IFI	50-348, 364/94-28-02	Closed	Evaluation of Settings for Westinghouse Unit 2 MOV 8811A Using the EPRI PPP Model (Section E8.2).
IFI	50-348, 364/94-28-03	Closed	Evaluation of Settings for Pratt Butterfly MOVs Using the EPRI PPP Model (Section E8.3).
LER	50-348, 364/95-06	Closed	Licensed Material Shipped to Incorrect Destination by Common Carrier (Section R8.1).
URI	50-348, 364/96-09-05	Closed	Failure to Search Contractor Trailer Prior to Entry Into the Protected Area (Section S8.1).

Discussed

Type	Item Number	Status	Description and Reference
IFI	50-348, 364/96-02-03	Open	Pre-action sprinkler system failures (Section F2.1).
URI	50-348, 364/96-09-08	Open	Adequacy of Kaowool qualification tests to scope installed configurations (Section F8.1).

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
ALM	Automated Laundry Monitor
Bhp	Brake Horsepower
BIT	Boron Injection Tank
BRT	Bottom of Rolled Transition
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CIR	Chemistry Incident Report
CO_2	Carbon Dioxide
DAD	Digital Alarming Dosimeter
DCP	Design Change Package
EDG	Emergency Diesel Generator
EPB	Emergency Power Board
EPRI	Electric Power Research Institute
EQ	Environmentally Qualified
ETP	Engineering Test Procedure
FCV	Flow Control Valve
FNP	Farley Nuclear Plant
HELB	High Energy Line Break
HP	Health Physics
HX	Heat Exchanger
IAW	In Accordance With
ID	Inside Diameter
IFI	Inspector Followup Item
IN	Information Notice
IP	Inspection Procedure
IPE	Individual Plant Examination
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Test
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOSP	Loss of Offsite Power
MCB	Main Control Board
MCR	Main Control Room
MOV	Motor Operated Valve
mrem	millirem
MS	Main Steam
NORB	Nuclear Operations Review Board
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OOS	Out of Service
OR	Occurrence Report
PA	Protected Area
PAHA	Post-Accident Hydrogen Analyzers
PCE	Personnel Contamination Event
PDR	Public Document Room
pH	The negative logarithm of the hydrogen concentration.

PPP	Performance Prediction Program
PPR	Piping Penetration Room
PRF	Penetration Room Filtration
QA	Quality Assurance
Radwaste	Radioactive Waste
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RE	Radiation Element
RG	Regulatory Guide
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RP&C	Radiological Protection and Chemistry
RPC	Rotating Pancake Coil
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
RxxCxx	SG tube location (e.g. R20C26 = Row 20 Column 26)
SAER	Safety Audit and Engineering Review
SBO	Station Blackout
SFP	Spent Fuel Pool
SG	Steam Generator
SI	Safety Injection
SNC	Southern Nuclear Operating Company
SOP	System Operating Procedure
SRO	Senior Reactor Operator
SS	Shift Supervisor
STP	Surveillance Test Procedure
SW	Service Water
TDAFW	Turbine Driven Auxiliary Feedwater
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TO	Tag Order
TS	Technical Specifications
U2RF11	Unit 2 eleventh refueling outage
UFSAR	Updated Final Safety Analysis Report
UOP	Unit Operating Procedure
URI	Unresolved Item
UT	Ultrasonic Testing
UTEC	Ultrasonic Examination
VIO	Violation
WO	Work Order
YTD	Year-to-Date