U.S. NUCLEAR REGULATORY COMMISSION (NRC)

REGION II

Docket Nos: 50-348 and 50-364

License Nos: NPF-2 and NPF-8

Report No: 50-348/98-03 and 50-364/98-03

Licensee: Southern Nuclear Operating Company (SNC)

Facility: Farley Nuclear Plant (FNP), Units 1 and 2

Location: 7388 North State Highway 95
Columbia, AL 36319

Dates: April 12 through May 30, 1998

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Enclosure 2

The NRC has concluded that information regarding the reason for Violation C. the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in LER 97-10-01. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the Farley Nuclear Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Atlanta, Georgia
this 1st day of July 1998

EXECUTIVE SUMMARY

Farley Nuclear Power Plant, Units 1 and 2
NRC Inspection Report 50-348/98-03, 50-364/98-03

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7-week period of onsite resident inspector inspection and announced inspections by regional inspectors.

Operations

- Control Room professionalism and communications remained good. Operating crew demeanor, team work and conduct were professional and effective. Operator attentiveness to Main Control Board (MCB) annunciator alarms and response to changing plant conditions were prompt. The operating crew consistently demonstrated a high level of awareness of existing plant conditions and ongoing plant activities. (Section O1.1)

- The inspectors concluded that the licensee adequately prepared for and then satisfactorily conducted Unit 2 midloop operations. (Section O1.3)

- The Unit 2 cycle 13 initial approach to criticality and restart were well-briefed, deliberate, and conservative. The reactor core performed within its design parameters. (Sections O1.5 and O1.6)

- The Unit 2 power ascension for Cycle 13, following the power uprate, was conducted in a safe and controlled manner. The unit achieved full power without a significant personnel incident or equipment problem. (Section O1.7)

Maintenance

- Maintenance and surveillance testing activities were generally conducted in a thorough and competent manner by qualified individuals in accordance with plant procedures and work instructions. Close coordination was maintained with the main control room during surveillance testing activities. (Section M1.1)

- The corrective actions for the March 28 and May 12 rod drop events were not thorough, but the corrective actions following the May 15, 1998 event appeared to be comprehensive and effective, pending completion of the licensee's root cause determination. (Section M1.4)

- A non-cited violation was identified for the licensee's failure to report a manual reactor trip in a timely manner. (Section M1.4)

- Inservice Inspection (ISI) activities were conducted in accordance with procedures and regulatory requirements. (Section M1.5)

- The Inservice Inspection/Nondestructive Examination (ISI/NDE) program lacked procedure qualification of high temperature liquid penetrant examination. (Section M1.5)

Engineering

- The inspectors concluded that the licensee had established suitable programmatic guidance to ensure that the regulatory requirements of 10 CFR 50.59 would be met by the various onsite and offsite organizations. However, the inspectors did identify several programmatic deficiencies and inconsistencies. Training of safety evaluation preparers and reviewers was adequate. (Section E1.1)

- Changes, tests and experiments were properly screened for 10 CFR 50.59 applicability, and adequately evaluated to ensure an unreviewed safety question (USQ) did not exist. Personnel preparing and reviewing safety evaluations were qualified. However, the documentation that addressed the USQ criteria in several safety evaluations lacked specificity and thoroughness. Furthermore, very few of the safety evaluation forms provided any direct evidence of a cross-disciplinary review. (Section E1.1)

- A violation was issued to the licensee because the original safety assessment for LER 97-10 was inadequate. In addition, the ability to safely shutdown and cooldown the plant from the HSDP was determined to have been in a degraded condition for about 12 years. This issue remains under NRC review and was identified as an apparent violation. (Section E8.1)

- A violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action was identified. The licensee identified three conditions adverse to quality of Control Room Ventilation System Functional System Design (FSD) Open Items, which were either inadvertently or inappropriately closed and not corrected. (Section E8.5)

Plant Support

- A weakness in exposure controls and poor communications contributed to the licensee exceeding its budgeted dose for the removal of Tri-Nuclear equipment and filters from the U2 lower reactor cavity due. (Section R1.1)

- For U2 Refueling Outage 12 (U2RF12) activities, dose expenditure exceeded original estimates due to expanded work scope, unexpected Residual Heat Removal (RHR) system maintenance problems, and elevated U2 Spent Fuel Pool dose rates. (Section R1.2)

- Worker Shallow Dose Equivalent (SDE) exposures resulting from personnel contamination events and work activities during the U2RF12 activities were evaluated properly and were within 10 CFR 20.1201 limits. (Section R1.2)

- Controls for minimizing workers' internal exposure during U2RF12 activities were effective. (Section R1.3)

- Respiratory protection training, fit tests, medical qualifications, and equipment status met 10 CFR 20.1703 requirements. (Section R1.4)

- Plant personnel observed working in the radiologically-controlled area (RCA) generally demonstrated appropriate knowledge and application of radiological control practices. (Section R2.1)

- The evaluated Radiation Monitor System (RMS) equipment was installed properly and the reviewed detector calibrations and functional tests were conducted in accordance with and met procedural, 10 CFR Part 20, and Offsite Dose Calculation Manual (ODCM) requirements. (Section R2.2)

- For 1997, program activities to control, monitor and document liquid and airborne radionuclide concentrations in effluents and in the offsite environment were implemented effectively. No significant environmental impact was identified. Projected offsite doses to the maximally exposed individual were a small fraction of ODCM and 40 CFR 190 specified limits. (Section R3.1)

- Extensive delays in returning a community particulate air sampler to service and lack of corrective actions to prevent recurrence was identified as a program weakness. (Section R3.1)

- The licensee Health Physics (HP) and Dosimetry (DOS) observation program continued to be implemented effectively and contributed to the reduced personnel errors observed for U2RF12 activities. (Section R7.1)

- Emergency Response Facilities (ERFs) were well-equipped and operationally ready to support an emergency response. Emergency response personnel were adequately trained and responded appropriately to a scheduled drill. (Sections P2.1 and P5.1)

- Changes to the Emergency Plan were made in accordance with 10 CFR 50.54(q). The emergency declaration on March 8, 1998, was made in accordance with the Emergency Plan. (Section P3.1)

- The 1996 and 1997 Emergency Preparedness (EP) program audits met the 10 CFR 50.54(t) requirement for an annual independent audit of the EP program. (Section P7.1)

- Security personnel activities observed during the inspection period were performed well. The security systems and barriers were adequate to ensure physical protection of the plant and complied with the Physical Security Plan. (Section S1.1)

- The failure to include a documented process in access control procedures for contractors to timely inform the Security Department of terminated individuals contributed to a violation for failure follow procedure to immediately terminate eight individuals' unescorted access. (Section S2.1)

- The licensee had in place a sound strategy that was capable of protecting vital equipment from acts intended to cause a significant release of radioactivity. (Section S4.1)

Report Details

Summary of Plant Status

Unit 1 operated continuously at 100% rated thermal power (RTP) for the entire inspection period, with the exception of several hours on May 21, when power was reduced to approximately 65%, due to a Steam Generator Feedwater Pump trip. The unit reached 360 days of continuous operation as of May 30, surpassing the previous Unit 1 record of 357 continuous power operation days.

Unit 2 was in a refueling outage for most of the inspection period. On May 15, operators attempted to restart Unit 2. However, the unit was manually tripped due to a dropped rod. After making repairs, Unit 2 was returned to criticality on May 17. The unit achieved 100% RTP on May 24 and continued to operate at this level through the end of the inspection period.

I. Operations

O1 Conduct of Operations

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

Following the guidance provided in Inspection Procedures (IPs) 71707 and 40500, the inspectors conducted frequent inspections of routine plant operations.

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

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

O1.2 Unit 2 Refueling (60710)

The resident inspectors observed Unit 2 refueling activities from the Main Control Room (MCR) and containment on April 27, 1998. The refueling was conducted in accordance with FP-APR-R12, "Refueling Procedure J.M. Farley Unit 2, Cycle XII - XIII Refueling," Revision (Rev.) 0. Refueling activities observed were performed in a well-controlled and methodical manner in accordance with procedures. Communications between the various stations were clear and concise. Personnel were familiar with the procedure and knowledgeable of the process and systems. No significant incidents occurred during fuel handling and all observed fuel assemblies were landed in their appropriate locations. The inspector concluded that fuel handling was accomplished in a professional and competent manner.

O1.3 Unit 2 Mid-loop Operations

a. Inspection Scope (71707)

The inspectors observed licensee preparations for establishing midloop conditions on Unit 2 in accordance with FNP-2-UOP-4.3, "Mid-loop Operations," Rev. 8. The inspector reviewed FNP-2-UOP-4.3, FNP-2-STP-18.4, "Containment Mid-Loop and/or Refueling Integrity Verification and Containment Closure," Rev. 17, and Generic Letter 88-17, "Loss of Decay Heat Removal," and verified selected portions of FNP-2-UOP-4.3, Section 2.0, Initial Conditions. The inspectors also performed MCR observations during midloop conditions to verify compliance with FNP-2-UOP-4.3.

b. Observations a Findings

The inspectors interviewed Operations and Training supervisors and determined that the operating crews had been provided basic midloop training during the first training cycle. In addition, the crew scheduled to initiate mid-loop conditions was to receive a detailed pre-evolution briefing prior to going into mid-loop operations.

The inspectors reviewed several procedures that were needed for mid-loop operation and found them to contain adequate information and appropriate detail to satisfy the concerns expressed in Generic Letter 88-17.

The inspectors observed MCR operations during mid-loop conditions that were established and maintained from May 3 through May 5. All required reactor vessel level indications were functioning properly and closely monitored by the operators. No significant problems were identified by the inspectors.

c. Conclusions

The inspectors concluded that the licensee adequately prepared for and then satisfactorily conducted Unit 2 mid-loop operations.

O1.4 Unit 2 Preparation For Startup and Mode Changes

The inspectors periodically reviewed FNP-0-SOP-103, "Return to Service Checklist," Rev. 10, and verified that mode-specific lists were up-to-date and complete. The inspectors verified that appropriate signoffs and reviews were completed for each mode change evolution.

O1.5 Unit 2 Startup and Initial Criticality

a. Inspection Scope (71711)

The inspectors observed initial criticality of Unit 2 for Cycle 13 and the commencement of zero power physics testing. The inspectors also observed the pre-evolution briefing.

b. Observations and Findings

On May 15, the inspectors attended the pre-evolution briefing for Unit 2 Cycle 12 initial criticality and low power physics testing. Because this was an infrequently performed evolution, the Engineering Support (ES) manager and test coordinator (i.e., nuclear engineer) conducted the briefing per FNP-0-AP-92, "Infrequently Performed Tests and Evolutions," Rev. 3. The briefing was attended by all affected parties and was comprehensive.

Unit 2 entered Mode 2 on May 15, when operators began to withdraw the control rod shutdown banks. An inspector monitored the approach to criticality during withdrawal of the control banks and subsequent Reactor Coolant System (RCS) boron dilution. The inspectors verified that the startup was performed in accordance with FNP-2-UOP-1.2, "Startup of Unit From Hot Standby To Minimum Load," Rev. 39, and FNP-2-STP-101, "Zero Power Reactor Physics Testing," Rev. 0.

The approach to criticality was conducted in a slow, deliberate manner in strict compliance with procedural instructions. Criticality was achieved within expected bounds of the estimated critical concentration (ECC) and predicted quantity of makeup water needed to dilute the RCS. All reactivity alterations were precisely controlled and directly communicated to the shift supervisor (SS) prior to implementing any change. The inverse count rate ratio (ICRR) was plotted methodically during the entire evolution and reflected control over reactor reactivity conditions by Operations and ES personnel. Overall, the Cycle 12 approach to criticality was performed well.

Later that evening, during the dynamic rod worth measurement on control bank 'D' with reactor power below the point of adding heat, control rod F10 dropped into the core from approximately 216 steps. Operators manually tripped the reactor and carried out applicable emergency operating procedures. A blown fuse was found on the movable gripper for rod F10. The licensee promptly notified the NRC of this event pursuant to 10 CFR 50.72 (b)(2)(ii). This was the third rod drop event for Unit 2 since March 28 (see Section M1.4).

c. Conclusions

The Unit 2 Cycle 13 initial approach to criticality was well pre-briefed, deliberate and conservative. The reactor core performed within its design parameters.

O1.6 Unit 2 Restart and Low Power Physics Testing

a. Inspection Scope (71711)

On May 17, the inspectors observed operators return the reactor to a critical condition by pulling rods and continuing with low power physics testing.

b. Observations and Findings

On May 17, after completing repairs to the rod control power cabinets (see Section M1.4), operators recommenced Unit 2 restart. An inspector monitored the approach to criticality during withdrawal of the control banks in accordance with FNP-2-UOP-1.2, "Startup of Unit From Hot Standby To Minimum Load," Rev. 39, and FNP-2-STP-101, "Zero Power Reactor Physics Testing," Rev. 0.

The approach to criticality was conducted in a slow, deliberate manner in strict compliance with procedural instructions. Criticality was achieved within the expected bounds of the estimated critical position (ECP). All reactivity alterations were precisely controlled and directly communicated to the SS prior to implementing any change. The inverse count rate ratio (ICRR) was plotted methodically during the entire evolution and reflected positive control over reactor reactivity conditions by Operations and ES personnel.

c. Conclusions

Overall, the Unit 2 Cycle 13 restart was well-controlled and the reactor core responded within design expectations.

O1.7 Unit 2 Power Ascension

a. Inspection Scope (71707)

From May 17 through 22, the inspectors observed portions of the Unit 2 power ascension and operations, as conducted by associated operating crews in accordance with FNP-2-UOP-3.1, "Power Operations," Rev. 38, and FNP-2-ETP-4441, "Power Ascension Following Unit Uprate," Rev. 0. In addition, the inspectors observed FNP-2-IMP-228.11, "NIS Power Range Channel N44 Current Rescale," Rev. 18, and portions of FNP-2-ETP-4440, "Steam Generator Water Level Control Test," Rev. 0, at the 33% power plateau.

b. Observations and Findings

The main generator was synchronized to the grid on May 18 and achieved full power on May 24. The Unit 2 power ascension and power operations were well-controlled and consistent with FNP-2-UOP-3.1 and FNP-2-ETP-4441 guidance. FNP-2-IMP-228.11 and FNP-2-ETP-4440 were conducted in a controlled, step-by-step manner and completed satisfactorily.

c. Conclusions

The Unit 2 power ascension for Cycle 13, following the power uprate, was conducted in a safe and controlled manner. The unit achieved full power without a significant personnel incident or equipment problem.

O2 Operational Status of Facilities and Equipment

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

General tours of safety-related areas were performed by the inspectors to observe the physical condition of plant equipment and structures, and to verify that safety systems were properly maintained and aligned. Overall material conditions for Unit 1 and Unit 2 structures, systems, and components (SSCs) were good, and safety-related system appeared to be properly aligned. Minor equipment and housekeeping problems identified by the inspectors during their routine tours were reported to the responsible SS and or maintenance department for resolution.

O2.2 Inspections of Safety Systems (71707)

Inspectors walked down the newly installed Unit 2 TriSodium Phosphate Baskets and recently repaired emergency core cooling system (ECCS) containment sump screens to verify operability. Accessible portions of these safety-related system components were verified to be adequately installed and in good operating condition. The inspectors did not identify any issues that adversely affected component operability.

O2.3 Unit 2 Containment Closeout Walkdown (71707)

On May 14, the inspectors toured the inside of Unit 2 containment shortly after entry into Mode 3. The inspectors identified a slight amount of debris which was removed prior to unit startup. A few minor leaks and housekeeping problems were reported to the Unit 2 SS for action. Overall, the licensee did an adequate job in cleaning and clearing out the containment.

02.4 Verification of Safety Tagging (71707)

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

- 98-0242-1, 1C Coolant Charging Pump Auxiliary Lube Oil Cooling Pump
- 98-1186-2, 2C Battery Charger
- 93-1843-2, Spent resin sluice pump
- 97-2742-2, Containment Spray Addition Tank
- 98-0467-2, Lower Equipment Room Air Handling Unit
- 97-2458-2, Turbine Driven Auxiliary Feedwater Pump

The inspectors verified that devices identified on the Tag Orders (TOs) were properly tagged and that the administrative aspects of filling out the tagging order forms were complete and correct.

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

O5 Operator Training and Qualification

O5.1 Unit 2 Power Uprate License Condition Training

On April 29, the NRC issued Amendment No. 137 to Facility Operating License (FOL) No. NPF-2 and Amendment No. 129 to FOL No. NRF-8 for FNP, Units 1 and 2, respectively. These license amendments authorized SNC to operate both units of FNP at reactor power levels up to 2775 megawatts-thermal (MWt), which was a power increase from the original license limit of 2652 Mwt. As part of the license amendments, the NRC approved certain new license conditions, one of which was that SNC shall complete classroom and simulator training regarding power uprate for operations crews on both units prior to the Unit 2 restart (i.e., before entering Mode 2) from U2RF12. The Unit 1 reference simulator was also to be temporarily modified to accommodate the training. On May 5, an inspector interviewed responsible training instructors and management to discuss the conduct of operator training pursuant to the new license conditions.

The inspector reviewed training lesson plan OPS-56202A, "Power Uprate," which addressed power uprate changes to: 1) System and Control Setpoints, 2) Technical Specifications, 3) Emergency Procedures, and 4) Accident Analysis. The inspector also reviewed applicable training attendance sheets to verify operator attendance for classroom and simulator training. Classroom training was held for all licensed reactor operators (RO) and senior reactor operators (SRO) during January through April 1998. Simulator training was conducted for the operating crews between April 22 and May 6, 1998. Based upon the interviews and document reviews, the inspector concluded that the Appendix C license conditions of Amendments Nos. 137 and 129, which required operator training prior to Unit 2 restart, were satisfactorily fulfilled.

O6 Operations Organization and Administration

O6.1 Peer Review by World Association of Nuclear Operators (WANO) (71707 and 40500)

The inspectors reviewed the Final Report of the "WANO Peer Review of Farley Nuclear Plant," conducted onsite during the month of July 1997. The inspectors' review of the Interim Report dated September 16, 1997, was documented in inspection report 50-348, 364/97-14. After reviewing the Final Report, the inspector concluded that there were no new safety issues identified which would require NRC follow-up action or reassessment of NRC perspectives regarding licensee performance.

O8 Miscellaneous Operations Issues

O8.1 Employee Concerns Program

a. Inspection Scope (40500)

The inspectors performed a review of a sample of Employee Concerns Program (ECP) files.

b. Observations and Findings

The licensee recently dedicated a full-time person to serve as ECP Coordinator to process employee concerns. This individual was a Shift Foreman and holds a Senior Reactor Operator (SRO) license. This individual has recently begun actively advertising the plant ECP and encouraging people to submit concerns.

The total concerns for 1995, 1996, and 1997 were 5. Concerns for 1998 year to date totaled 30. A review of all ECP packages for 1995, 1996, and 1997 and 4 ECP packages for 1998 found two that had followup commitments. However, there was no documentation indicating that these commitments were completed.

One concern stated that there were no searches of individuals entering or exiting the contractor parking area. Security had committed to review their procedures. Due to communications problems, security reviewed the entrance/exit procedures for the wrong area. Consequently, no actions were taken. ECP personnel acknowledged the error when the inspectors questioned them about the disposition of this concern. Upon subsequent review, the licensee stated that no actions were required because searches were conducted prior to entering the protected area. Also, the licensee had independently implemented random exit searches.

The other concern was associated with inconsistent verification of controlled leakage when swapping a charging pump. The licensee had committed to change the Unit Operating Procedures (UOPs) and the System Operating Procedures (SOPs). This commitment was to be entered into an informal tracking system for procedure enhancements. The SOP changes were entered into the tracking system and incorporated, but the UOP changes were not entered. Consequently, these changes were not made. However, the licensee planned to incorporate the changes during the next revision cycle of the procedure.

c. Conclusions

There has been a significant increase in usage of the Employee Concerns Program during 1998. For the ECP packages reviewed, one commitment was not entered into the tracking system.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance and Surveillance Testing Activities (61726 and 62707)

Using the guidance provided in IP 61726 and IP 62707, the inspectors observed and reviewed portions of selected licensee corrective and preventive maintenance activities, and routine surveillance testing including detailed reviews of the following:

- FNP-2-STP-40.0, "Safety Injection with Loss of Off-Site Power Test," Rev. 29
- FNP-2-STP-40.3, "Phase A Isolation Test," Rev. 1
- WA485614, Replace 1A Condensate Pump bearing
- FNP-1-STP-109.1, "Power Range Neutron Flux Channel Calibration," Rev. 10
- FNP-0-MP-7.3, "Turbine Driven Auxiliary Feedwater Pump Overspeed Trip Setpoint Checks," Rev. 4

During the observation of FNP-0-MP-7.3, the inspectors noted several unsuccessful attempts to perform the maintenance. Mechanics were unable to adjust the Turbine-Driven Auxiliary Feed Water (TDAFW) overspeed trip setpoint within the required tolerances. The source of the problem was narrowed down to non-equivalent parts that were replaced on the overspeed trip mechanism during the Unit 2 Refueling Outage 12 (U2RF12). The licensee, in consultation with the vendor, concluded that a part of the overspeed device was "custom fit" at the factory and that the off-the-shelf component would not work. The licensee determined that the entire overspeed device would be purchased and factory tested in the future. After the original parts were put back into the overspeed trip mechanism, a successful test was accomplished the following day.

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

M1.2 Unit 2 Safety Injection with Loss Of Offsite Power Test

The inspectors observed the preparations for and performance of FNP-2-STP-40.0, "Safety Injection with Loss of Off-Site Power Test," Rev. 29C, on April 25. The inspectors also observed portions of the follow-up testing on April 29, and reviewed the completed test package.

The test was satisfactorily completed with the exception of five valves, inadvertently omitted due to personnel error, and several relatively minor exceptions, which were noted and rescheduled. While recording post-test component positions, the licensee identified that five valves were not placed in the required pre-test alignment because the operator performing the pretest lineup missed page 5 of Table 2. The licensee initiated Occurrence Report (OR) 2-98-162 to evaluate this occurrence. These valves, and the known test exceptions, were retested on April 29. The inspectors observed the retest and verified that all exceptions from the initial test were included. The test and retests adequately verified SI operation with a loss of offsite power.

M1.3 Residual Heat Removal (RHR) Heat Exchanger Head Gasket Replacement

The licensee replaced the RHR heat exchanger (HX) head gaskets, under Work Orders (WOs) M00203005 and M00168359, to eliminate small borated water leaks. The job was complicated when seven of the studs which were threaded through the tube sheet stuck and had to be cut out with specialized equipment. This significantly delayed completion of work and resulted in the dose for the job being almost double the budgeted dose.

On May 1, a contractor noted that some of the studs (one on each HX endbell and multiple studs on the inlet and outlet flanges) did not have complete thread engagement with the nuts. In most cases, only one or two threads were not engaged. The American Society of Mechanical Engineers (ASME) Code required full thread engagement. The condition was missed by the craftsmen, the craftsmen's supervision, and inspection personnel. The licensee issued OR 2-98-171, evaluated the condition for current plant conditions, initiated a formal root cause investigation, and installed longer studs where needed for full thread engagement. Based on initial calculations, the licensee determined that for the current plant conditions (i.e. Modes 5 and 6), the RHR HXs would meet their intended function with the as-found thread engagement. Therefore, based on having adequate bolt strength but not meeting the ASME Code, the licensee considered this to be a degraded but operable condition.

On May 4, an inspector walked down other systems to determine if inadequate thread engagement was a problem on other safety-related systems. The inspector identified limited examples of potentially inadequate thread engagement on the 1A Containment Spray pump supply and discharge flanges, 1C CCP discharge flange, and the 2B RHR pump casing. After the walkdown, the inspector asked the Plant Modifications and Maintenance Support (PMMS) manager if licensee personnel planned to walkdown any other safety-related systems for thread engagement discrepancies. The manager stated that no further system walkdowns were planned but that he would discuss the issue with senior plant management. The inspector then presented his walkdown findings to the licensee for resolution. On May 5, the licensee commenced walkdowns to identify thread engagement problems on other safety-related systems. Based on the initial calculations for the RHR HX thread engagement and the as-designed safety margins, the inspector concluded that this was not a significant safety concern. This issue is identified as IFI 50-348, 364/98-03-01, Inadequate Thread Engagement, pending inspector review of the licensee's walkdown results and evaluations.

M1.4 Dropped Rod During Rod Operability Testing

a. Inspection Scope (62707)

Inspectors observed control rod troubleshooting and reviewed the applicable occurrence reports and Licensee Event Report (LER), associated with the dropped control rods.

b. Observations and Findings

On May 12, control rod K-2 (control bank A) was dropped during rod operability testing. This was the same rod that dropped on March 28 during a scheduled Unit 2 refueling outage shutdown (refer to IR 98-02) when a fuse associated with the stationary gripper coil blew. At that time, the licensee's troubleshooting concluded that there was an intermittent problem in the rod control power cable that crossed from the reactor cavity to the reactor head. The suspected cables were replaced during the outage. However, for the May 12th event, a fuse associated with the movable gripper was found to be blown. In both cases, operators manually tripped the subcritical reactor. The licensee suspended further control rod operability testing and commenced troubleshooting to identify the problem. Licensee personnel subsequently concluded that the movable gripper coil fuse had experienced a late failure due to the prior overcurrent condition that blew the stationary gripper coil fuse. Maintenance replaced the movable gripper coil fuse and exercised the bank several times.

Following replacement of the fuse, the inspectors observed Instrument and Control troubleshooting activities (e.g., electrical power current traces) of each control rod while operators exercised applicable banks.

The licensee did not report the May 12 event to the NRC until May 15. Failure to report the manual reactor trip within four hours as required by 10 CFR 50.72(b)(2)(ii) constituted a violation. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1 of the NRC Enforcement Policy. This is identified as NCV 50-364/98-03-02, Failure to Report Manual Reactor Trip in a Timely Manner.

On May 15, another control rod dropped during zero power physics testing (see Section O1.5). The licensee's NSSS supplier reviewed the event. A potential concern was identified regarding suitability of fuse types for the Control Rod Drive Motor Generator sets and the control rod system. The licensee had on occasion installed a different type of fuse than recommended by the vendor, considered to be equivalent by the licensee. All of the Unit 2 control rod drive fuses (approximately 160) were replaced with the vendor's recommended fuses and the reactor was restarted. The inspectors will continue to review the root cause determinations and the associated LERs. The licensee's corrective actions appeared to have been successful and the repetitive dropped rod problem corrected, pending results of the on-going Root Cause Investigation.

c. Conclusions

The corrective actions for the March 28 and May 12 rod drop events were not thorough, but the corrective actions following the May 15, 1998 event appeared to be comprehensive, pending completion of the licensee's root cause determination.

A non-cited violation was identified for the licensee's failure to report a manual reactor trip in a timely manner.

M1.5 Inservice Inspection Unit 2

a. Inspection Scope (73753)

To evaluate the licensee's inservice inspection (ISI) program and the program's implementation, the inspectors reviewed procedures, observed work in progress, and reviewed selected records. Observations were compared with applicable procedures, the Updated Final Safety Analysis Report (UFSAR), and ASME B&PV Code Sections V and XI, 1989 Edition, No Addenda (89NA).

Specific areas examined included the following observations: magnetic particle (MT) examination of Item Nos. APR2-4350-20 and APR1-1300-S35; manual ultrasonic (UT) examination of Item No. ARP2-4350-20; visual (VT-1) examination of Item No. APR1-4303-QV021(B); visual (VT-3) examination of Item No. APR2-4613-SS-12297; and data acquisition activities associated with eddy current (ET) examinations of steam generator (S/G) tubing. The inspectors reviewed selected completed examination reports and procedures.

Procedures reviewed included: FNP-0-NDE-100.1, "Measuring and Recording Techniques for NDE Examinations," Rev. 2.; FNP-0-NDE-100.5, "Liquid Penetrant Examination (Color Contrast and Fluorescent)," Rev. 4; FNP-0-NDE-100.11, "Magnetic Particle Examination," Rev. 3; FNP-0-NDE-100.21, "Visual Examination VT-1," Rev. 1; FNP-0-NDE-100.22, "Visual Examination VT-2," Rev. 2; FNP-0-NDE-100.23, "Visual Examination VT-3," Rev. 2; FNP-0-NDE-100.32, "Qualification of Ultrasonic Instruments," Rev. 2; FNP-0-NDE-100.31, "Manual Ultrasonic Examination of Full-Penetration Welds (0.200 to 6.0 Inches)," Rev. 4, with TCN 4A; FNP-0-NDE-100.34, "Manual Ultrasonic Examination of Welds in Vessels," Rev. 6; FNP-0-NDE-100.35, "Ultrasonic Thickness Examination Procedure," Rev. 1; FNP-0-NDE-100.37, "Manual Ultrasonic Examination of Reactor Coolant Pump Flywheels," Rev. 2; FNP-0-NDE-100.38, "Manual Ultrasonic Examination of Nozzle Inner Radius," Rev. 2; FNP-0-NDE-100.39, "Manual Ultrasonic Examination of Bolts and Studs Greater than 2 inches in Diameter," Rev. 3; FNP-0-NDE-100.40, "Manual Ultrasonic Examination of Centrifugal Charging Pump Case," Rev. 1; and FNP-0-NDE-100.41, "Manual Ultrasonic Examination of Cast Stainless Steel Pipe Welds," Rev. 1, with TCN 1A.

The inspectors performed an independent evaluation of indications to confirm the licensee's ISI examiners' evaluations. In addition, the inspectors conducted an independent VT-3 inspection of the following supports previously examined by the licensee to confirm their results: APR1-4301-2HR-R155 and APR2-4619-SS-12459.

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

The inspectors observed activities associated with insertion and expansion of S/G tube sleeves.

b. Observations and Findings

Unit 2 S/G tubing was subjected to ET examination. This examination was planned to include: bobbin - 100% full length; +Point rotating pancake (RPC) 100% top of tubesheet (TTS)± 3-inches hot leg, 20% TTS ± 3-inches cold leg, row 1 U-bends S/G 2A, row 2 U-bends S/G 2C, and all bobbin indications. For the alternate repair criteria program, the ET examinations were to include all tube support plate (TSP) bobbin indications over 2 volts, all dents over 5 volts, and large mixed residuals (33 in each S/G). Due to finding 2 tubes with indications in the S/G 2C cold leg TTS +Point, the +Point program was expanded to 100% of the inservice tubes. Due to finding an outside diameter stress corrosion cracking (ODSCC) indication at one TSP intersection in S/G 2B, that had a large mixed residual signal, the +Point program was expanded to look at the next 66 largest residuals. No indications were identified in the expanded sample of 66.

Procedure FNP-0-NDE-100.5, Rev. 4, Appendix A-2, provided step by step instructions for the performance of PT examinations in the temperature range 145 °F to 325 °F, using the following Sherwin Doubl-Chek® visible solvent removable family: KO-17 Penetrant, D-350 Developer, and KO-19 Cleaner. ASME B&PV Code Section V, paragraph T-647 requires procedure qualification for PT examinations that are to be conducted outside if the range of 60 °F to 125 °F. The licensee indicated that at present, it did not have any of the high temperature penetrant consumable materials. The inspectors determined that no examinations had been conducted in accordance with FNP-0-NDE-100.5, Appendix A-2. The licensees's approval and issuance of a PT procedure for examinations outside of the range of 60 °F to 125 °F, without first performing a qualification in accordance with T-647, was considered an inadvertent omission of the licensee's ISI/NDE programs.

The inspectors observed several personal safety concerns regarding improperly secured ladders and personnel working more than six feet above the floor without proper hand rails or safety harnesses. The inspectors reported these observations to the licensee who took immediate corrective actions to address these issues.

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

c. Conclusion

The Inservice Inspection/Nondestructive Examination (ISI/NDE) program lacked procedure qualification of high temperature liquid penetrant examination.

III. Engineering

E1 Conduct of Engineering

E1.1 10 CFR 50.59 Safety Evaluation Program And Implementation

a. Inspection Scope (37001)

By letter dated December 2, 1997, the licensee submitted Revision (Rev.) 14 to the Updated Final Safety Analysis Report (UFSAR) for the time period of November 5, 1995, to June 4, 1997. This letter also included a Summary Report of all changes, tests, and experiments (CTEs) that were completed under the provisions of 10 CFR 50.59 over the same period. The licensee's December 2 summary included approximately 135 changes.

The inspectors conducted a review of the licensee's program for meeting the regulatory requirements of 10 CFR 50.59 and examined its implementation. The inspectors reviewed applicable administrative and controlling procedures, training materials, and numerous safety evaluations, including associated UFSAR Rev. 14 changes. In addition, the inspectors attended a meeting by the Plant Operations Review Committee (PORC) that included review and approval of several safety evaluations. The inspectors also reviewed recent audits conducted in the area of 10 CFR 50.59 safety evaluations.

b. Observations and Findings

Program Review

The inspectors reviewed onsite Administrative Procedure (AP) FNP-0-AP-88, "Nuclear Safety Evaluations," Revs. 2 and 3, and corporate Farley Nuclear Project procedure GO-NG-42, "50.59 Evaluations," Rev. 4. In addition, the inspectors reviewed Nuclear Engineering Procedure (NEP) 8-102, "Preparation Of Safety Evaluations," Revs. 6 and 7, and Nuclear Engineering Procedure Instruction (NEPI) 4-0, "Design Change Packages," Rev. 2. The principal offsite design organizations, Southern Company Services (SCS) and Bechtel, used NEP 8-102 and NEPI 4-0 for conducting safety evaluations of plant design changes. At the time of the inspection, the licensee was transitioning from using NSAC-125, "Guidelines For 10 CFR 50.59 Safety Evaluations," to NEI 96-07, "Guidelines For 10 CFR 50.59 Safety Evaluations," dated September 1997. Revision 2 of FNP-0-AP-88, Rev. 6 of NEP 8-102, and Rev. 2 of NEPI 4-0 were based on NSAC-125; the other procedures were recent revisions to endorse NEI 96-07. The licensee has committed to fully implement NEI 96-07 by June 30, 1998.

After reviewing the above procedures, the inspectors concluded that the licensee had established suitable programmatic guidance to ensure that the regulatory requirements of 10 CFR 50.59 would be met by the various onsite and offsite organizations. However, the inspectors did identify several program inconsistencies, as described below:

1) The onsite, corporate, and design organization transition from NSAC-125 to NEI 96-07 was poorly coordinated. At the time of the inspection, onsite personnel were using FNP-0-AP-88, Rev. 2, based on NSAC-125 (Rev. 3 was approved but not issued); whereas, corporate project personnel were using GO-NG-42 based on NEI 96-07 since January 15, 1998. Also, offsite design organizations were required to use two guidelines at the same time with one based on NSAC-125 (i.e., NEPI 4-0) and another based on NEI 96-07 (NEP 8-102, Rev. 7).

2) Although the offsite design organizations conducted the majority of all actual safety evaluations (i.e., addressing the unreviewed safety question (USQ) criteria), the level of detail of their procedural guidance was minimal compared to the onsite and corporate project organizations. NEPI 4-0 lacked much of the general and specific guidelines contained in FNP-0-AP-88 and GO-NG-42. [However, this was previously recognized by the licensee who was preparing to implement a new, more detailed instruction PDI 5.8-102, "Preparation of Safety Evaluations (10 CFR 50.59)," based on NEI 96-07.]

3) Lack of written guidance for addressing the necessity of reverifying safety evaluations for changes that are not implemented after many months or years.

4) Conduct of cross-disciplinary preparation/review of safety evaluations was not addressed by 10 CFR 50.59 program procedures. Only FNP-0-AP-1, Rev. 36, "Development, Review, and Approval of Plant Procedures," makes any direct reference to cross-disciplinary reviews of safety evaluations and even that applies only to the 10 CFR 50.59 screening. Onsite and offsite design change control procedures were vague and unclear regarding cross-disciplinary reviews of 10 CFR 50.59 safety evaluations.

5) Definition and explanation of safety margin in FNP-0-AP-88, Rev. 2, was inconsistent with NRC Inspection Manual Part 9900: 10 CFR Guidance issued April 1996 as "10 CFR 50.59 Interim Guidance on the Requirements Related to Changes to Facilities, Procedures and Tests (or Experiments)," and revised in October 1997. [Note,

inspectors did not review NEI 96-07, or licensee NEI 96-07 related procedures, against the NRC interim guidance.]

6) Responsibilities of the Manager, described in FNP-0-AP-88, Revs. 2 and 3, for approving safety evaluations that do not pass the 10 CFR 50.59 screening, are not addressed.

10 CFR 50.59 Screening Process

The inspectors reviewed 20 completed safety evaluation forms for CTEs that the licensee determined did not satisfy the requirements for performing an evaluation of the USQ criteria of 10 CFR 50.59. For these safety evaluation forms, only Section B, "10 CFR 50.59 Applicability," was filled out, which determined that a 10 CFR 50.59 safety evaluation was not required. The 20 CTEs selected were screened for 10 CFR 50.59 applicability during the time period from August 1997 to April 1998. The inspectors did not identify any CTE that was improperly screened for 10 CFR 50.59 applicability.

Safety Evaluations - USQ Criteria

The inspectors selected about two dozen safety evaluations of safety-significant CTEs that were determined to be 10 CFR 50.59 applicable to verify that each of the individual safety evaluation preparers/reviewers/approvers were qualified to conduct these evaluations. The inspectors also selected 17 safety evaluations of safety-significant CTEs for a detailed review on the completeness and adequacy of the answers to the USQ criteria of 10 CFR 50.59. All of the CTEs selected included a variety of systems and different engineering disciplines. However, they were almost exclusively related to plant design changes, requests for engineering assistance (REAs), and as-built notifications (ABNs). Very few procedure changes ever met the 10 CFR 50.59 applicability determination, and the licensee rarely performed tests and experiments not described in the UFSAR. Of the safety evaluations reviewed, the majority were performed by offsite design organizations.

Unlike the FNP site and corporate project, the inspectors found that the offsite design organizations did not maintain any lists of qualified preparers/reviewers/approvers, but rather relied on the individual engineering supervisors and managers to keep track of who was qualified in their areas of responsibility. This practice made it very difficult for the inspectors to independently verify the qualifications of personnel from offsite organizations. Consequently, the inspectors had to rely on licensee assurance that offsite design personnel were qualified, based on their review of individual personnel files. The inspectors did verify that FNP site and corporate project personnel were qualified to conduct and review the selected safety evaluations. During this effort, the inspectors also observed that very few of the safety evaluation forms provided any evidence of a cross-disciplinary review which was consistent with the lack of programmatic guidance in this area. In response to this observation, the licensee maintained that cross-disciplinary reviews of safety evaluations would normally be covered during the design input review, and verification. However, the inspectors determined that neither NEP 4-1, "Establishing Design Input Requirements," Rev. 4, for preparing design input records or NEP 4-9, "Design Verification," Rev. 4, for conducting design verifications provided any explicit guidance for multi-discipline reviews of safety evaluations. Consequently, even after reviewing numerous design change packages (DCPs) associated with the selected safety evaluations and reflecting on NEP 4-1 and 4-9, the inspectors were unable to conclude that safety evaluations were specifically receiving multi-discipline reviews. Failure to perform these reviews may be a contributing factor to the lack of necessary detail in safety evaluations as discussed below.

Of the 17 safety evaluations reviewed in detail for adequacy and completeness of their answers, the inspectors did not identify a CTE that involved a USQ. However, many of the safety evaluations provided a minimal or insufficient level of detail in answering the questions to address the 10 CFR 50.59 USQ criteria. In general, the information contained in the "Background and Description" portions of Section A, "Activity Summary," of the safety evaluation form tended to be quite detailed. Also, the responses to the questions of Section B, "10 CFR 50.59 Applicability," were suitable. But the answers to the questions in Section C, "USQ Criteria," were typically very summarized and lacked specificity. For several of the safety evaluations, the Section C answers were so brief and generalized that, by themselves, they would have been inadequate. However, in almost all of these cases, the reader was able to obtain sufficient information from the description in Section A to satisfy the appropriate question of Section C. The major problems with this approach were that it made reviewing the safety evaluation more difficult, suggested that the preparer did not understand the scope of each question, and was inconsistent with the NSAC-125 and NEI 96-07 guidance for providing complete and thorough answers to the seven questions addressed by the descriptive information.

Some particular examples of safety evaluations that provided inadequate detail in Section C to address the USQ criteria of 10 CFR 50.59, but where the information could basically be found in Section A, were as follows:

- 10 CFR 50.59 Evaluation, Rev. 5, for DCP 96-0-9012-2-006;
- 10 CFR 50.59 Evaluation, Rev. 3, for DCP 97-0-9182-0-004;
- 10 CFR 50.59 Evaluation for DCP B-97-1-9192-0-003;
- 10 CFR 50.59 Evaluation for ABN 95-0-0589; and,
- 10 CFR 50.59 Evaluation, Rev. 1, for DCP 95-2-8932-1-004.

In addition to these safety evaluations, there was an example where neither Section A nor the answers to Section C of the 10 CFR 50.59 safety evaluation form provided enough detail to determine that a USQ did not exist. However, in this instance, the licensee was able to demonstrate to the inspectors that, even though the safety evaluation forms lacked the information needed to answer the questions for determining a USQ, there was sufficient documented basis in the associated DCP and/or references listed in Section A to determine that a USQ did not exist. This example was the 10 CFR 50.59 Evaluation for DCP 95-2-8932-1-004.

10 CFR 50.59 does not specify the manner in which a safety evaluation should be documented. As such, failure to provide sufficiently detailed answers to the "seven questions" in Section C does not specifically constitute a noncompliance with 10 CFR 50.59. However, the guidance in NEP 8-102 clearly states that " the safety evaluation should be written as a stand-alone [emphasis added] document with sufficient detail." It also states that "a thorough description is required because other personnel reviewing the documentation may not be familiar with the physical plant." There was nothing in NEP 8-102 (nor FNP-0-AP-88 or GO-NG-42) to suggest that the answers of Section C could rely upon information in Section A, the DCP package, or references in order to address all elements of the subject change that could reasonably affect a USQ determination. Quite the contrary, program guidance recommended completeness and specificity. Adequate documentation to address the USQ criteria is considered a weakness in the implementation of the licensee's 10 CFR 50.59 program.

10 CFR 50.59 Summary Report Descriptions

The inspectors compared 13 summary descriptions of CTEs reported to the NRC pursuant 10 CFR 50.59 in the December 2, 1997, letter to the description of changes contained in the actual 10 CFR 50.59 safety evaluations. Inspection report (IR) 96-07 had identified examples in the licensee's previous 10 CFR 50.59 report to the NRC that were either incomplete, did not clearly identify the nature of the change, or used plant-specific acronyms that were not readily recognizable. During this review, the inspectors did not identify any of these examples and concluded that the descriptions of changes contained in the most recent 10 CFR 50.59 summary report were complete and adequately described the change.

UFSAR Changes Resulting From CTEs

The inspectors reviewed ten design changes identified in the 10 CFR 50.59 summary report and compared them to the actual changes contained in Rev. 14 to the UFSAR. No discrepancies were identified.

Training Associated with 10 CFR 50.59 Program Activities

The inspectors reviewed the licensee's training program, contained in the Farley Technical Staff and Management document TSM-510, "Nuclear Safety Evaluations," July 1996, and associated training material used by the training department. The training material included: (1) personnel requirements for performing 10 CFR 50.59 evaluations; (2) an FSAR overview; (3) an FSAR search program use; (4) examples of 10 CFR 50.59 evaluations and screening material for 10 CFR 50.59 applicability; (5) 10 CFR 50.59 evaluation guidelines; (6) NSAC-125 and guidance related to its use; and (7) administrative procedure FNP-0-AP-88, Rev. 2.

The inspectors also reviewed FNP-98-0067-TRN, "Designation of Qualified Reviewers," which contains a matrix of personnel that are qualified to prepare and/or review 50.59 safety evaluations. Individuals were selected for 10 CFR 50.59 training based on need and department managers' recommendations. The training department maintained a list of trained individuals and the next due date for refresher training. The inspectors verified that those individuals listed in FNP-98-0067-TRN have maintained their training current. However, the inspectors noted that the FNP-0-AP-88 requirement that, "Personnel who prepare, review, or approve 10 CFR 50.59 evaluations will be trained every two calendar years," was inconsistent with the guidance of Technical Staff and Managers Curriculum Guide for TSM-510. The licensee's actual practice of retraining conformed most closely with the curriculum guide rather than FNP-0-AP-88. Although there is no specific 10 CFR 50.59 requirement for refresher training, the licensee was informed of the conflict between TSM-510 and FNP-0-AP-88.

Prior to November 1997, qualifications to perform 10 CFR 50.59 evaluations were based on maintaining training current and successful completion of the one-day course on TSM-510. However, since that time, individuals that attend the TSM-510 course were given a written multiple choice exam at the end of the course. Approximately 150 of 300 people listed in FNP-98-0067-TRN, have taken the written exam. The licensee has indicated that the remaining individuals will be given a written exam when refresher training is taken.

During the transition to NEI 96-07, Rev. 0, "Guidelines for 10 CFR 50.59 Safety Evaluations," and the process of revising FNP-0-AP-88, the training department was also updating associated 10 CFR 50.59 training material as applicable. In anticipation of the NEI 96-07 transition and to provide onsite and offsite safety evaluation preparers/reviewers with more in-depth and comprehensive training, the licensee contracted for a special one-day training course, primarily during the Summer and Fall of 1997. The SNC 10 CFR 50.59 Evaluation Training Program of "Meeting the 10 CFR 50.59 Evaluation Challenge - A Program to Achieve Excellence," was given to all qualified preparers/reviewers. The inspectors reviewed the training manual used and found it to be comprehensive and thorough.

Audits of 10 CFR 50.59 Program Implementation

Very few audits of the 10 CFR 50.59 program and its implementation have been performed. Historically, any audits of licensee safety evaluation activities were typically included as part of broader audits of other programs (e.g., design change control). However, an audit of SNC-Farley Support-Nuclear Engineering and Licensing was conducted during July 1997 that included implementation of GO-NG-42, Rev. 3. Although no audit finding reports were identified, one comment was made regarding lapsed training for certain individuals. Also, during the team inspection, a site-specific spot audit was in progress "to obtain the status of, as well as determine, the degree of consistency in implementation of NEI 96-07." The inspectors reviewed the audit report and audit notes associated with these audits. Both audits were of very limited scope and detail. The overall paucity of auditing in this area provided the inspectors with insufficient information to conclude that the licensee's audit program was effectively assessing conformance with 10 CFR 50.59. However, the inspectors did note that routine auditing of 10 CFR 50.59 activities were not specifically required by the audit program as defined by TS, UFSAR, and Operations Quality Assurance Policy Manual (OQAPM).

c. Conclusion

The inspectors concluded that the licensee had established sufficient programmatic guidance to ensure that the regulatory requirements of 10 CFR 50.59 would be met by the various onsite and offsite organizations. However, the inspectors did identify several programmatic deficiencies and inconsistencies. Training of safety evaluation preparers and reviewers was adequate.

Changes, tests and experiments were properly screened for 10 CFR 50.59 applicability, and adequately evaluated to ensure an unreviewed safety question did not exist. Personnel preparing and reviewing safety evaluations were qualified. However, the documentation that addressed the USQ criteria in several safety evaluations lacked specificity and thoroughness. Furthermore, very few of the safety evaluation forms provided any direct evidence of a cross-disciplinary review.

Licensee audits of the 10 CFR 50.59 program were few in number and very limited in scope and detail.

E8 Miscellaneous Engineering Issues (IP 92903)

E8.1 (Closed) Unresolved Item (URI) 50-348,364/98-01-04, Inadequate Safety Assessment For Mis-wired Hot Shutdown Panel MOVs

(Closed) Licensee Event Report (LER) 50-348,364/97-10-01, Motor Operated Valve Local - Remote Control Circuit Wiring Discrepancies

a. Inspection Scope

The inspectors reviewed revised LER 97-10-01, applicable Abnormal Operating Procedures (AOP), licensee's response to a request for engineering assistance (REA), and conducted walkdowns of the equipment in question.

b. Observations and Findings

In May 1997, the licensee discovered that the control power circuitry for five motor-operated valves (MOVs) were mis-wired. The mis-wired condition defeated the electrical isolation fuse design that ensured that these five MOV's would be operable from the Hot Shutdown Panel (HSDP) during a fire in the MCR or cable spreading room. The licensee promptly repaired the improper wiring and submitted LER 97-10 for operating in a condition outside of the design basis.

As referenced in Section E8.1 of IR 98-02, 10 CFR 50.73(b)(3) requires the licensee to assess the safety consequences and implications of reportable events, including the availability of other systems or components that could have performed the same function as that which had failed. After reviewing the safety assessment of the original LER 97-10, the inspectors concluded that the licensee failed to perform an adequate assessment of the safety consequences and implications of the mis-wired MOV's during a fire in the MCR or cable spreading room that would necessitate implementing the unit-specific AOP-28.1, "Fire or Inadvertent Fire Protection System Actuation in the Cable Spreading Room," or AOP-28.2, "Fire in the Control Room." Furthermore, the licensee failed to describe availability of other systems, components, or manual actions to compensate for the loss of MOV functions. Failure to perform an adequate safety assessment constitutes a violation of 10 CFR 50.73(b)(3) and is identified as VIO 50-348, 364/98-03-04, Inadequate Safety Assessment for Mis-wired Hot Shutdown Panel MOVs. Based on this violation, URI 50-348,364/98-01-04 is closed. The licensee revised its original LER, especially the safety assessment, to provide a better understanding of the risk and safety significance of the reportable event. Additionally, no new corrective actions for the event were identified. The inspectors reviewed the revised safety assessment and concluded it adequately addressed the safety consequences and implications of the event, as well as, described the availability of other systems, components, or manual actions to compensate for the loss of the MOVs.

The licensee's revised safety assessment determined the possible adverse impacts of postulated plant fires in the cable spreading rooms (CSR) and the main control room (MCR) upon the ability to shutdown and cooldown the plant considering the mis-wired MOV's. For these limiting fires, the consequences of inadequate electrical isolation of these five mis-wired MOVs on the HSDP and MCB are discussed below.

Power Operated Relief Valve (PORV) Block Valves

Postulated fires in the MCR and CSR could have resulted in both mis-wired PORV block valves becoming inoperable from the HSDP. However, the licensee concluded that a failed open PORV could be manually de-energized and thereby closed to re-establish the RCS pressure boundary, by using procedural guidance previously approved by the NRC in a SER for an Appendix R exemption. The inspectors verified that this procedural guidance was available at the Hot Shut Down Panel.

Component Cooling Water (CCW) to Secondary Heat Exchanger Isolation Valve MOV 3047 and Refueling Water Storage Tank (RWST) to Charging Pump Suction Isolation Level Control Valves (LCVs) 115B and 115D

The more significant concern identified by the licensee's safety assessment was the potential that a MCR or CSR fire coincident with a LOSP could have resulted in a LOCA due to loss of cooling to the RCP seals. [Loss of cooling to the RCP seals would require a loss of seal injection and CCW flow to the thermal barrier heat exchangers.]

Under certain fire-induced failure conditions, the centrifugal charging pump (CCP) suction could be lost, resulting in possible vapor binding and damage to the operating CCP. In addition, a fire-induced spurious valve closure could isolate CCW supply flow to the RCP thermal barrier. Restoration of CCW flow or charging flow would then require manual actions because of the loss of control of MOV 3047, and LCV-115B and LCV-115D, from the HSDP due to blown MCR fuses. According to the licensee's assessment, the redundant CCPs would have been available to start from the HSDP, but only after manually opening the RWST to CCP suction MOVs (and venting the CCPs, as necessary) to reestablish seal cooling through seal injection. Similarly, CCW flow could only be restored by manual operation of MOV 3047. However, these manual actions, without specific operator training or procedures, would have significantly delayed restoration of seal flow.

The inspectors walked down the MCR and CSR wiring for the components of concern to determine the probability of a fire affecting both the CVCS (letdown, CCPs, VCT discharge valves) and CCW MOV 3047. The inspectors found that MOV 3047 and the CVCS were on the same section of the MCB, approximately 12 feet apart. The respective cables dropped vertically from the switches through floor penetrations directly to the CSR. The majority of the cables in that section of the MCB had a braided stainless steel jacket. There were no vertical separators in the MCB to provide physical isolation. Once in the CSR, the cabling split north and south to the respective "A" and "B" train cable trays. The inspectors concluded, based on the physical layout, that it would take a major MCR or CSR fire to affect both charging flow and MOV 3047 to cause a loss of RCP seal cooling and subsequent seal LOCA.

During the 12-year period that the affected MOVs were mis-wired, no fires or other significant plant events occurred that necessitated taking control of the MOVs at the HSDP.

The licensee's assessment concluded that it would be extremely unlikely that a fire of sufficient magnitude to adversely impact components as described above to occurr. The licensee considered it even more unlikely that a fire of this magnitude could occur coincident with an LOSP. A significant fire in the main control room was not considered by the licensee to be a credible event because it is continuously occupied, power circuits are minimized, and combustibles are limited. The CSRs have automatic fire suppression systems, fire alarms, and limited combustible materials. The CSRs are also equipped with manually actuated low pressure carbon dioxide fire suppression systems. The licensee also considered that the advance of a postulated fire would not have been instantaneous and components adversely affected by the fire would not have been affected simultaneously. Operators in the MCR would have been alerted to component malfunctions either through indications, alarms, or procedural steps.

The licensee concluded that while the MOVs were mis-wired, the ability to achieve safe shutdown for certain postulated plant fires coincident with a LOSP was degraded. This possible loss of capability to shutdown and cooldown the plant from outside the MCR, as required by 10 CFR 50, Appendix R [illegible] is identified as an apparent violation, EEI 50-348, 364/98-03[illegible], HSDP Loss of Function.

In addition to the corrective actions of LER 97-10-00, the licensee reported in LER 97-10-01 that the Abnormal Operating Procedures for responding to fires in the CSR or MCR were revised. The revised procedures now require operators (if time allows) to open LCV-115B and LCV-115D prior to evacuating the control room in order to minimize the potential for losing suction to the operating CCP. This action was considered an enhancement to the procedure by the licensee. These procedure changes were verified by the inspectors. The revised LER 97-10-01 is considered closed.

c. <u>Conclusion</u>

A violation was identified because the original safety assessment for LER 97-10 did not completely address the safety consequences and implications of the possible failure of five mis-wired motor-operated valves at the Hot Shutdown Panel during a control room or cable spreading room fire. The subsequent supplemental LER did provide sufficient information for an adequate safety assessment of the event. In addition, an apparent violation was identified due to the determination that the licensee's ability to safely shutdown and cooldown the plant from the HSDP was in a degraded condition for about 12 years.

E8.2 (Closed) VIO 50-348, 364/97-11-04, Failure to Implement a Test Program for Service Testing of the TDAFW Battery

The licensee responded to this VIO in correspondence dated December 17, 1997, and initiated Corrective Action Report (CAR) 2321. The inspectors reviewed the licensee's written response and completed CAR, verified implementation of corrective actions, and reviewed the test data for the Unit TDAFW battery service test. This VIO is closed.

E8.3 (Discussed) VIO 50-348, 364/97-11-03, TDAFW Battery Installation and Check Valve Test Deficiencies

The licensee responded to this VIO in correspondence dated December 17, 1997, and initiated CAR 2322. This CAR was not complete at the end of this inspection period. The inspector verified that the TDAFW Battery racks were rebuilt per the applicable drawings. This VIO will remain open pending review of the completed CAR and the check valve test deficiencies corrective action.

E8.4 Modification of Penetrations for GL 96-06

The inspectors verified that relief valves were installed in Penetration 30, Pressure Relief Tank (PRT) Makeup, and Penetration 31, Reactor Coolant Drain Tank (RCDT) Drain, per the licensee's letter dated May 23, 1997, in response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions."

E8.5 (Closed) URI 50-348, 364/98-01-05, Failure to Track and Correct Conditions Adverse to Quality

(Closed) IFI 50-348, 364/98-01-06, Control Room Ventilation Testing

a. Inspection Scope (37551)

The inspectors reviewed a variety of tracking list data and closure documentation, interviewed personnel, and walked down the systems.

b. Observations and Findings

The subject of Licensee Event Report (LER) 97-13, "Operating Outside the Design Basis Due to Control Room Exhaust Isolation Dampers Not Closed," originated from Open Item CRV-007, identified during the Control Room Ventilation (CRV) Functional System Description (FSD). Historically, the plant had operated with the CRV exhaust dampers open. But during a self-assessment of the CRVS, questions arose regarding the design basis requirements for positioning these exhaust dampers and FSD CRV-007 was opened. In April 1995, a design evaluation was performed offsite to address CRV-007. This evaluation concluded that the CRVS emergency pressurization system was designed to operate with these dampers shut. However, the results of the evaluation were misinterpreted at the site and the dampers were procedurally left open. In August 1997, during the FSAR verification process, the open damper position was again questioned by the licensee. Licensee personnel determined that the plant had operated outside its design basis while the exhaust dampers were open. These dampers were then shut and LER 97-13 was issued (see IR 98-01, Section E8.2)

As part of the corrective actions for LER 50-348, 364/97-13, the licensee conducted a review of all the closed out "Open Items" previously identified during the CRV FSD and Safety System Self-Assessment (SSSA) and ascertained that 5 of 19 open items had been closed out without any evidence that the recommended corrective actions were implemented. Responsibility for two of these items (Open Items CRV-010 and CRV-019) had been assigned to the site.

Open Item CRV-010 originally identified that some areas surrounding the main control room (MCR) could be pressurized to greater than 0.125 inches water gauge (w.g.), thus allowing unfiltered in-leakage, greater than assumed, into the MCR. Bechtel letter AP-21274, dated June 7, 1995, completed the evaluation and identified two spaces where single failures could cause a room adjacent to the MCR to be pressurized greater than 0.125 inches W.G. This letter provided recommendations to resolve the concern of over-pressurizing areas next to the Control Room and thereby not allow greater than assumed unfiltered in-leakage into the MCR. These recommendations were provided to the site from corporate engineering via a letter (NEL 95-0189), dated July 6, 1995. The Bechtel letter also recommended closing item CRV-FSD-010, "Control Room Pressurization from Adjacent Areas," and it was removed from the corporate tracking list. However, during this time the open item was also inadvertently removed from the site tracking list. Later in November 1997, during the review of completed CRVS, FSD, and SSSA Open Items as a corrective action for LER 97-13, the licensee determined that no evidence (e.g. revised procedures, etc.) could be found to ascertain that the recommendations for CRV-FSD-010 had ever been implemented or dispositioned at the site. This item was subsequently reopened, and the proposed recommendations were still being evaluated at the end of this inspection period.

Open Item CRV-019 was originally concerned with weaknesses in testing the pressurization system to support the allowable open penetrations in the control room boundary. During the SSSA of the CRV system, Assessment Observation CRV-MECH-02, dated November 17, 1995 (later designated as Open Item CRV-019) identified some potential weaknesses in the testing to support MCR allowable open penetrations calculation. The response to CRV-MECH-02 (NEL 96-0069, dated February 27, 1996) addressed several specific issues, but did not look at the larger issue of MCR boundary degradation. During corrective actions for LER 97-13, former "Open Items" from the FSD/SSSA were reviewed to verify adequate closure. The licensee identified that the response to close out this issue did not address the boundary degradation issue (NEL 97-0526, dated December 19, 1997) and recommended that periodic testing be accomplished to assure that the MCR penetration opening allowance was conservative.

On February 2, 1998, the licensee established an administrative Limiting Condition of Operation (LCO) that prevented additional breaches in the control room boundary. However, on March 5, 1998, corporate engineering sent an evaluation (NEL 98-0088) to the site which recommended against testing to validate the pressurization performance, but did state that the plant may want to consider adding a test to determine boundary degradation. In addition, the letter re-issued a Bechtel letter which confirmed that the 21.21 square inch opening was conservative. At that time the restriction on additional boundary breaches was lifted. However, no additional boundary breaches were required. After discussions with the licensee, Engineering Support conducted a test on March 25, 1998, for the CRV system that included air flow data. The licensee's calculation determined that only 10 square inches of opening could be allowed and still maintain the required MCR over-pressure. The licensee re-instituted an administrative LCO to prevent CR boundary breaches greater than the new calculated area. On April 21, 1998, the licensee added a temporary change to FNP-0-AP-16, "Conduct of Operations - Operations Group," Rev. 27, which removed the 21.21 square inch administrative limit and referenced a data sheet in the Plant Curve Book, which is updated quarterly to reflect the boundary degradation of the control room.

Until this time, the testing being done did not verify that the control room minimum pressure could be maintained with a 21.21 square inch opening, the licensee's administrative limit.

An inspector's review of the most recent surveillences, FNP-0-STP-26.2, "Control Room Pressurization/Filtration Operability Test" for 'A' and 'B' Trains," Rev. 12, indicated that the system can maintain the minimum control room pressure in its current condition and that the licensee is cognizant of the need to maintain the control room boundary integrity.

Open Items CRV-007, CRV-010, and CRV-019 were conditions adverse to quality that were not adequately corrected. In each of the three previously described cases, the licensee had originally identified a deficiency and then either inadvertently or inappropriately closed it out. The licensee then re-identified the items and either has taken or is completing corrective actions on the individual issues. Failure to adequately correct conditions adverse to quality is identified as VIO 50-348, 364/98-03-06, Inadequate Corrective Actions for MCR

Ventilation System. Also, the inspectors expressed concern to the licensee that FSD open items for systems in addition to the CRVS may have been closed with inadequate corrective actions. By the end of this inspection period, the licensee had not reviewed or reverified the adequacy of the corrective actions closed out for other FSD systems.

c. Conclusions

A violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action was identified. The licensee identified three conditions adverse to quality of Control Room Ventilation System Functional System Design (FSD) Open Items, which were either inadvertently or inappropriately closed and not corrected.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls (83750, 84750, 86750)

R1.1 Radiological Controls (83750)

a. Inspection Scope

Radiological controls associated with ongoing Unit 1 (U1) routine operations and with Unit 2 Refueling Outage 12 (U2RF12) activities were reviewed and evaluated by the inspectors. Reviewed program areas included area cleanliness and housekeeping, area postings, radioactive material and waste (radwaste) container labels, high and locked-high radiation area controls, and procedural and radiation work permit (RWP) implementation. The inspectors made frequent tours of the Radiologically Controlled Areas (RCAs) and directly observed worker and Health Physics Technician (HPT) performance during selected tasks.

Established Radiation Protection (RP) program guidance and implementation were compared against commitments detailed in the Updated Final Safety Analysis Report (UFSAR), and in procedural, Technical Specification (TS), and 10 CFR Part 20 requirements.

b. Observations and Findings

High and locked high radiation area controls were established and maintained in accordance with TS requirements. Area postings and container labels were proper for the radiological conditions and met procedural, TS, or 10 CFR 20 Subpart J requirements. Improvements were noted in labels provided for containers of radioactive materials. Contamination and radiation surveys were conducted in accordance with procedural requirements. Radiation and contamination survey results met established regulatory and procedural limits.

On April 29, 1998, the inspectors observed work activities and HP practices associated with removal of Tri-Nuclear equipment and filters from the lower reactor cavity conducted in accordance with Specific RWP No. 298-2491, Revision (Rev.) 1. During completion of the task, a remote handling tool was damaged, causing the dose expenditure to escalate to approximately 985 millirem (mrem), exceeded the budgeted dose of 900 mrem. The following poor radiological practices, job planning weaknesses, and communication issues were identified and discussed with licensee representatives:

- Mock-up training was not provided for removing the spent filters from the vacuum system and for transferring the material to the Unit 2 drumming room. Design differences between the previous and current vacuum system model, which now required a specific alignment of the filters within their housings for proper removal, was not identified. Improper alignment during the initial attempt to remove a filter from the vacuum system resulted in excess force being applied and the remote handling device being damaged.

- Methods and controls to limit personnel exposure were minimally effective. Plans for use of an extension tool while transferring the filters in containment was abandoned after problems were identified during removal of the initial filter from the vacuum system; extensive exposure time was required to manually tie and untie ropes to bags used to hoist the eight vacuum system filters from the lower to the upper cavity area; and on several occasions, workers entered designated exclusion areas during transfer of the filters. Also, when a supplemental teledose monitor, provided to a Health Physics Technician (HPT) handling a spent filter, alarmed as a result of an improper dose rate alarm setting, the individual returned to transfer an additional filter prior to change-out of the alarming unit.

- Planning and communication weaknesses were identified during a post-job briefing and from followup discussions with participating operations, maintenance, HPT and "As Low As Reasonably Achievable" (ALARA) staff. For example, maintenance workers were knowledgeable of general area dose rates associated with the filters, but were not fully aware of the significant hazard from the filter contact dose rates and the importance of using remote tools in handling the filters. Furthermore, previous maintenance staff safety concerns regarding use of the remote tool to remove the filters from the vacuum equipment and potential contamination concerns from ropes used to suspend the vacuum equipment in the cavity were not incorporated into planning for the current task.

c. Conclusions

Radiological controls were established and maintained in accordance with procedural, TS, and 10 CFR 20 Appendix J requirements.

A weakness in exposure controls and poor communications contributed to the licensee exceeding its budgeted dose for the removal of Tri-Nuclear equipment and filters from the U2 lower reactor cavity due.

R1.2 External Exposure (83750)

a. Inspection Scope

The inspectors discussed and reviewed deep dose equivalent (DDE) and shallow dose equivalent (SDE) exposures to workers involved in U2RF12 activities. Personnel contaminations, documented as personnel contamination events (PCEs), i.e., dispersed contamination greater than or equal to (≥) 5000 disintegrations per minute per 100 square centimeters (dpm/100cm^2) and specks ≥ 100000 dpm/probe areas, were reviewed and discussed.

Dose assessment methods and assumptions, where applicable, were reviewed for technical adequacy. Dose results were compared against 10 CFR Part 20 limits.

b. Observations and Findings

Estimated dose data, as measured by Digital Alarming Dosimeter (DAD) for U2RF12 activities, were reviewed and discussed with responsible staff. As of April 30, 1998, dose expenditure for outage activities, approximately 169.064 person-rem, exceeded the original projected dose expenditure of 155.956 person-rem. The licensee identified problems with Residual Heat Removal (RHR) pump maintenance activities, expanded scope of mid-loop valve maintenance, and unexpected elevated dose rates in the U2 spent fuel pool (SFP), contributing to the elevated person-rem expenditures. From review of selected Occurrence Reports and discussions with licensee staff, the inspectors verified that RHR maintenance and SFP dose expenditure issues were being reviewed and evaluated.

As of April 29, 1998, approximately 29 personnel contamination event reports were documented with only one event requiring a skin SDE determination. For the affected individual, a hot particle located on the upper right forearm resulted in an assigned shallow dose equivalent (SDE) of approximately 7.76 rem. Licensee assumptions and details regarding physical location, length of exposure and isotopic characteristics of particle were appropriate. The inspectors noted that all assigned doses were within 10 CFR 20.1201 limits.

c. Conclusions

For U2RF12 activities, dose expenditure exceeded original estimates due to expanded work scope, unexpected maintenance problems, and elevated U2 spent fuel pool dose rates.

Worker SDE exposures resulting from personnel contamination events and work activities during the U2RF12 activities were evaluated properly and were within 10 CFR 20.1201 limits.

R1.3 Internal Exposure (83750)

a. Inspection Scope

Results of selected investigative whole-body count (WBC) analyses conducted during the U2RF12 outage were reviewed in detail.

b. Observations and Findings

From review of WBC analysis records of workers' positive radionuclide intakes, the inspectors identified one individual whose initial WBC analyses data resulted in an assigned committed effective dose equivalent (CEDE) exceeding 10 mrem. The inspectors noted that as of April 29, 1998, approximately 12 investigative WBC analyses were conducted as a result of specific events, usually documented in Radiation Worker Performance Observations, which could cause or indicate potential radionuclide intakes resulting in internal exposure. The estimated maximum intake was approximately 158 nanocuries (nCi), resulting in an assigned CEDE of 12 mrem. The inspectors verified that the 12 mrem CEDE was added to the DDE to provide the total effective dose equivalent (TEDE) documented in the individual's official exposure records. No other evaluated worker intakes exceeded 10 mrem, i.e., 0.2 percent of the annual limit of intake (ALI) required to be documented by licensee procedures.

c. Conclusions

Controls for minimizing workers' internal exposure during U2RF12 activities were effective.

R1.4 Respiratory Protection (83750)

a. Inspection Scope

Respiratory protection program implementation for U2RF12 activities was reviewed and evaluated. The review verified training, fit testing, and medical qualifications for selected licensee and contractor personnel who were supplied and used respiratory protection equipment.

Licensee activities were reviewed and evaluated against procedural and 10 CFR 20.1703 requirements.

b. Observations and Findings

Workers using respiratory protective equipment during U2RF12, were fit tested, medically qualified, and trained in accordance with procedural requirements.

c. Conclusions

Respiratory protection program implementation for U2RF12 activities met established procedural and 10 CFR 20.1703 requirements.

R2 Status of Radiological Protection and Chemistry Controls Facilities and Equipment

R2.1 Radiologically Controlled Area (RCA), Units 1 and 2 (71750)

Overall cleanliness of the RCA remained good. Plant personnel observed working in the RCA generally demonstrated appropriate knowledge and application of radiological control practices. Health physics technicians generally provided positive control and support of work activities in the RCA.

R2.2 Radiation Monitoring Systems

a. Scope

Design and calibration issues were reviewed and discussed for selected Radiation Monitoring System (RMS) sampling equipment and detectors. Design issues for the RE-29B, Plant Vent Post-Accident Vent monitor were reviewed. Calibration activities for the U2 Containment High Range Monitor (CHRM) RE-27A were observed and resultant calibration data were reviewed and discussed. In addition, design issues associated with effluent stream flow pathways for the RE-29B particulate sampler were reviewed and verified.

Installed equipment was evaluated against recommendations specified in American National Standards Institute (ANSI) N13.1-1969, American National Standard Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities. Calibration activities were evaluated against applicable sections of the Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS), and Offsite Dose Calculation Manual (ODCM) requirements. In addition, calibration activities to meet a March 14, 1983 Order to implement and maintain licensing commitments associated with Three Mile Island (TMI) Action Item II.F.1 for the CHRMs special calibrations were reviewed.

b. Observations and Findings

The installed U1 RE-29B RMS sample line flow path was found to be acceptable. The U2 CHRMs electronic calibrations and functional tests and isotopic calibration checks were conducted in accordance with Surveillance Test Procedure FNP-2-STP-227.18, "In-Containment High Range Radiation Area Monitor R-27A," Rev. 9, and Radiation Control and Protection (RC&P) Procedure FNP-2-RCP-272, "Isotopic Calibration Check of the Unit 2 Containment High Range Area Monitors," Rev. 3. The calibrations were conducted by electronic signal substitution for all range decades above 10 Roentgens per hour (R/hr). No regulatory issues were identified for the test and calibration data reviewed.

c. Conclusions

The evaluated RMS equipment was installed properly and the reviewed detector calibrations and functional tests were conducted in accordance with and met procedural, 10 CFR Part 20, and ODCM requirements.

R3 Radiation Protection and Chemistry Documentation (84750)

R3.1 Radiological Effluent and Environmental Monitoring Reports

a. Inspection Scope

Data and conclusions documented in the 1997 Annual Radiological Environmental Operating Report and the 1997 Annual Radioactive Effluent Release Report were reviewed and discussed. The contents and conclusions of the reports were evaluated against the applicable sections (§§) of TSs 6.8 and 6.9.1, and § 7 of the Offsite Dose Calculation Manual (ODCM).

b. Observations and Findings

The inspectors verified that the 1997 Radiological Environmental Operating Report was prepared and submitted in accordance with TS and ODCM requirements. Based on trend data for radionuclide concentrations in offsite environmental matrices at control and indicator stations, no discernible offsite effects or trends were demonstrated from plant effluent discharges to the environment. The licensee properly determined the controlling receptor to evaluate the maximum dose to a member of the public beyond the site boundary based on releases and current land-use census data. From review of the 1997 environmental monitoring program sampling deviations required by ODCM Section 7.1.2.4, the inspectors noted that community airborne particulate monitoring station number (No.) 1108 was inoperable from approximately November 18, 1997, through January 27, 1998, due to construction at the electric substation which supplied power to the equipment. Farley Nuclear Plant (FNP) Occurrence Report No. 973135, generated in response to finding the power off on November 25, 1997, initially documented that power would be interrupted for approximately two weeks, but an attached note indicated that as of January 13, 1998, power had not been restored to the sampling equipment. Corrective actions to prevent recurrence, such as the use of portable samplers or securing another source of electrical power within the immediate vicinity was not addressed. The inspectors identified the lack of detailed corrective actions to prevent recurrence of the extensive out-of-service condition due to a known power supply interruption, extending past the original two-week estimate, as an environmental monitoring program weakness.

The 1997 Annual Radioactive Effluent Release Reports was submitted in accordance with TS and ODCM requirements. For the report period, calculated offsite doses from liquid and gaseous effluent releases were a small fraction of the ODCM limits.

c. Conclusions

For 1997, program activities to control, monitor and document liquid and airborne radionuclide concentrations in effluents and in the offsite environment were implemented effectively. No significant environmental impact was identified. Projected offsite doses to the maximally exposed individual were a small fraction of ODCM and 40 CFR 190 specified limits.

Extensive delays in returning a community particulate air sampler to service and lack of corrective actions to prevent recurrence was identified as a program weakness.

R7 Quality Assurance in RP&C Activities

R7.1 Licensee Self-Assessment Activities (83750, 84750, 86750)

a. Inspection Scope

The inspectors reviewed implementation and status of the licensee's Health Physics Observation program. Program implementation and results were evaluated against commitments initially documented in an October 25, 1996 licensee response to a Notice of Violation regarding improper dosimetry use.

b. Observations and Findings

The inspectors noted that observations of both Health Physics (HP) and Dosimetry (DOS) practices continued. The observed data were assessed for the most current 1000 observations made and sorted into 26 separate types of poor practices or issues. Each identified item was subsequently assigned to one of twenty-three separate work groups. Results routinely were presented to upper management and workers. For identified error rates exceeding five and fifteen percent of the HP and DOS practices observed, Occurrence Reports were initiated and additional corrective actions taken. From review of the licensee weekly observation trend data for the U2RF12 outage, the inspectors noted that the percent errors for both HP and DOS issues were reduced by approximately 50 percent relative to weekly data collected during the previous Unit 1 Refueling Outage 14 (U1RF14) activities. For the U2RF12, the majority of identified issues were associated with incorrect radioactive material handling, violation of HP boundaries, spread of contamination, and incorrect dress out.

c. Conclusions

The licensee HP and DOS observation program continued to be implemented effectively and contributed to the reduced personnel errors observed for U2RF12 activities.

R8 Miscellaneous RP&C Issues (83750, 84750)

R8.1 (Closed) IFI 50-348,364/98-01-07: Review Licensee Actions to Improve Radioactive Material Container Label Effectiveness.

In response to inconsistencies and poor practices noticed for radioactive material container label types and information required by 10 CFR 20.1501, the licensee had assigned a senior HPT responsibility to review and provide oversight of the subject program area. Based on improvements in the radioactive material container labeling program activities noted during the current inspection period, this item is closed.

P2 Status of Emergency Preparedness (EP) Facilities, Equipment, and Resources

P2.1 Facility Inspection

a. Inspection Scope (82701)

The inspectors examined the licensee's emergency response facilities (ERFs) and equipment to assess their adequacy and to determine whether they were maintained in a state of operational readiness as specified in the Farley Emergency Plan.

b. Observations and Findings

The inspectors toured the Control Room, Technical Support Center (TSC), Operational Support Center (OSC), Emergency Operations Facility (EOF), and the alternate EOF. Selected equipment, supplies, and communications systems within these facilities were inspected. All tested equipment and systems were found to be in operable condition. The facilities were well-maintained.

c. Conclusions

ERFs were well-equipped and operationally ready to support an emergency.

P3 Emergency Preparedness (EP) Procedures and Documentation

P3.1 Emergency Plan

a. Inspection Scope (82701)

The inspectors reviewed recent revisions to the Emergency Plan to determine whether changes were made in accordance with 10 CFR 50.54(q), and plan implementation. In addition, the implementation of the plan of March 8, 1998, was reviewed.

b. Observations and Findings

The current revision of the Emergency Plan was administrative in nature. The Emergency Plan was implemented on March 8, 1998, with a Notification of Unusual Event (NOUE) due to high river water level. There was a partial TSC activation and review of documentation revealed that the required notifications were completed in a timely manner.

c. Conclusions

Changes to the Emergency Plan were made in accordance with 10 CFR 50.54(q). The NOUE on March 8, 1998, was made in accordance with the Emergency Plan.

P5 Staff Training and Qualification in EP

P5.1 Training of Emergency Response Personnel

a. Inspection Scope (82701)

The inspectors evaluated the training program for the Emergency Response Organization (ERO) through review of program documentation and observation of licensee training functions.

b. Observations and Findings

The licensee conducted a program of periodic integrated response drills (typically six per year) to enhance the training for ERO personnel. In an effort to gauge the effectiveness of the emergency response training program, the inspectors observed a previously scheduled ERO training drill on May 21. ERO personnel activated the ERFs in a timely manner and responded capably to the simulated emergency, which included event classifications of Alert, Site Area Emergency, and General Emergency. Minor problems with the ERO's response efforts were identified by licensee drill monitors for corrective action. The inspectors also observed an EOF tabletop drill on May 19 involving real-time setup of the facility and a round-table discussion of staff functions and interfaces.

c. Conclusions

The conduct of regular integrated drills enhanced the quality of ERO training. Drill monitors effectively identified response problems for corrective actions. ERO personnel were adequately trained and responded appropriately to a simulated event.

P6 EP Organization and Administration

P6.1 EP Program Organization

a. Inspection Scope (82701)

The inspectors reviewed this area to determine if changes in personnel had occurred which could adversely affect the management and implementation of the EP program.

b. Observations and Findings

The organization of the EP program was reviewed and discussed with licensee management representatives. Two changes to the EP organization were noted. The position of Emergency Planning Technician was reassigned in September 1997 to an individual who had previously been a member of the radiation protection group at Farley. This individual's professional development included a one-week training course in EP in December 1997.

A new Emergency Management Director for Houston County was recently appointed. According to licensee management representatives, this change had not had a negative impact upon the working relationship between the licensee and Houston County. The inspectors were informed that no other significant changes in management personnel for offsite interface/support agencies had occurred during the past two years.

c. Conclusions

No degradation had occurred in the EP program since the previous inspection.

P7 Quality Assurance in EP Activities

P7.1 10 CFR 50.54(t) Audit of Emergency Preparedness Program

a. Inspection Scope (82701)

The inspectors reviewed this area to assess the quality of the required annual audit of the emergency preparedness program, and to verify that the audit met the requirements of 10 CFR 50.54(t).

b. Observations and Findings

The inspectors reviewed documentation associated with the following EP program audits conducted by the licensee's Quality Assurance (QA) group:

- Safety Audit and Engineering Review (SAER) Audit of Farley Nuclear Plant-Emergency Planning Report No. 97-EP/16.

- SAER Audit of Farley Nuclear Plant-Emergency Planning Report No. 96-EP/16-1.

The audits were thorough and independent, and the nature of the identified issues indicated inclusive understanding of the EP area by the auditors. The audits provided evidence of the licensee's ability to self-identify emergency preparedness program issues.

c. Conclusions

The 1996 and 1997 EP program audits met the 10 CFR 50.54(t) requirement for an annual independent audit of the EP program.

P7.2 Effectiveness of Licensee's Corrective Action Program for EP Issues

a. Inspection Scope (82701)

The inspectors reviewed this area to evaluate the licensee's program for identifying, tracking, and resolving problems in emergency preparedness.

b. Observations and Findings

The licensee formally identified and tracked EP issues by means of the "Emergency Planning Punchlist." The licensee's list of open EP items is used to track all substantive findings, including many improvement items derived from drill critiques and carried at the lowest priority. Although the punchlist was maintained by the Emergency Planning Coordinator and was not integrated with any plant-wide tracking system, it was periodically distributed for updating by the assigned group for each item. This method was effective for resolving identified EP deficiencies and issues.

c. Conclusions

The licensee's program for identifying, tracking, and resolving problems in EP was effective.

P8 Miscellaneous EP Issues

P8.1 (Closed) Inspector Follow-up Item (IFI) 50-348, 364/96-14-01: Exercise Weakness--Significant Emergency Information Was Not Communicated to the Appropriate Emergency Manager in a Timely Manner.

This exercise weakness from the 1996 full-participation exercise was identified because significant emergency information was not communicated to the appropriate emergency manager in a timely manner. The training drill observed by the inspectors on May 21, 1998, provided the opportunity for the inspectors to focus on the transfer of emergency information from the interim Emergency Director (ED) in the simulator to the ED in the TSC, and later from the ED to the Recovery Manager in the Emergency Operations Facility. In all cases, the transfer of information was done clearly with repeat-backs to assure understanding, and it was done timely. This item is closed.

S1 Conduct of Security and Safeguards Activities

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

The inspectors verified that selected portions of site security program plans were being adequately implemented. Disabled vital area doors were properly manned and controlled. Security personnel activities observed during the inspection period were performed well. Site security systems were adequately maintained and functional to ensure the physical protection of the plant. However, the inspectors did identify two minor instances in which Security personnel were not attentive to equipment problems that adversely impacted effectiveness of physical security barriers: 1) Inoperative MCR door card reader green light (contrary to plant policy egress was allowed without verifying green light) and 2) Broken door latch on bullet hardened door outside PAP (door was blocked open, rather than disabling latch and leaving door shut. Although not specifically addressed by the Physical Security Plan (PSP), these barriers were in a degraded condition without compensatory measures in place. Once notified, Security promptly resolved each instance.

S2 Status of Security Facilities and Equipment

S2.1 Protected and Vital Area Access Control

a. Inspection Scope (81700)

The inspectors reviewed the PSP to determine if the licensee's access control program for personnel and packages met the commitments specified therein.

b. Observations and Findings

On April 21, the inspectors observed a reliability test conducted at the Primary Access Portal (PAP). A disabled weapon was placed inside a lunch cooler that was carried by a licensee individual with unescorted access. The search officer immediately discovered the weapon and detained the individual. The officer located in the final access control booth appropriately locked the protected area turnstiles and summoned assistance.

The inspectors reviewed records for 17 favorably terminated individuals with respect to the licensee's action to remove unescorted access. Procedure FNP-0-SP-11, "Badging Procedures," Rev. 13, required that changes in personnel access requirements caused by termination of employees will be reported immediately to the Security Site Manager. The necessary action will be taken to remove the individuals' name from access. In addition, procedure FNP-0-AP-42, "Access Control," Rev. 26, Section 7.5.3, required that individuals' names be removed from the appropriate access list immediately upon termination of need.

Of the 17 records reviewed, the inspector determined that 8 of the individuals did not have their unescorted access removed from the security computer ranging from 1 to 11 days after the individual had been terminated. All eight individuals had access to protected and vital areas; however, no individual accessed those areas after termination. Security removed access upon notification.

The inspector determined that although procedures did exist, clarification was needed as to contractors' responsibilities. Neither procedure had a process in place to ensure that contractor personnel who no longer needed unescorted access were immediately removed from the security computer. The inspector reviewed a Change Order (CO) for a contractor which was currently providing work at FNP and found that the CO simply stated to follow access procedures.

The failure to immediately report terminations of 8 employees to the Security Department is identified as Violation 50-348, 50-364/98-03-07, Failure To Promptly Terminate Security Access.

c. Conclusions

The licensee was appropriately searching individuals and packages prior to entrance into the protected area. The failure to include a documented process in access control procedures for contractors to timely inform Security of terminated individuals contributed to a violation for failure to immediately terminate 8 individuals' unescorted access.

S2.2 Protected Area Detection

a. Inspection Scope (81700)

The inspectors evaluated the licensee's protected area detection capability to determine if provisions of Section 5.3 of the PSP were met.

b. Observations and Findings

On April 22, the inspectors observed the licensee test two perimeter zones. Both zones alarmed appropriately. The inspector additionally performed a walkdown of the perimeter and determined that the design, placement, and coverage of the intrusion detection system met the requirements specified in the PSP.

c. Conclusions

A test of two perimeter zones identified that they alarmed appropriately. A walkdown of the perimeter intrusion detection system identified that design, placement, and coverage met the requirements of the PSP.

S2.3 Protected and Vital Area Barriers

a. Inspection Scope (81700)

Section 3 of the PSP outlined protected and vital area barriers that are in place at FNP. The inspectors evaluated those barriers to ensure that the criteria were being met.

b. Observations and Findings

The inspector performed a walkdown of protected and vital area barriers. Fences and gates were intact and met the overall height requirement. Manholes were appropriately secured and isolation zones were free and clear to assure a distinct field of vision. Protected area barriers were separated from vital area barriers.

Vital area barriers were appropriately in place and contained no openings greater than 96 square inches. Vital area doors were locked and alarmed. Access was controlled by a security force member, card reader, or the Central/Secondary Alarm Stations. The inspector accompanied a security officer who performed vital area door checks as part of his post. All vital area doors were secured and alarmed in the Central and Secondary Alarm Stations when opened. Vital area penetration points were secured by locks, alarms, or welded in place.

FNP-0-SP-30, "Declassification of Vital Area/Systems/Equipment," Rev. 0, dated March 11, 1994, was reviewed by the inspector. The licensee devitalized four valve boxes during Mode 5 of the outage. This devitalization of equipment met procedural requirements.

c. Conclusions

Protected and vital area barriers were appropriately placed, maintained, and secured as specified in Section 3 of the PSP. The licensee followed procedure to devitalize equipment during Mode 5 of the outage.

S3 Security and Safeguards Procedures and Documentation

S3.1 Security Program Plans

a. Inspection Scope (81700)

To determine if requirements were met, the inspectors reviewed Rev. 8 of the Training and Qualification Plan, which was submitted under 10 CFR 50.54(p).

b. Observations and Findings

Revisions to the Training and Qualification Plan met the requirements of 10 CFR 50.54(p). Administrative changes and clarification statements were also noted.

c. Conclusions

A revision to the Training and Qualification Plan did not decrease the effectiveness of the plan and met the requirements of 10 CFR 50.54(p).

S4 Security and Safeguards Staff Knowledge and Performance

S4.1 Response Capabilities

a. Inspection Scope (81700)

The inspectors reviewed and evaluated the licensee's response force strategy to determine if the licensee was capable of engaging an adversary force to preclude penetration of vital area barriers and any act intended to cause a significant release of radioactivity.

b. Observations and Findings

The inspector reviewed the Security Response Plan, Rev. 5, and drill critiques for the last two quarters. Three of the four response teams participated in the drills and the fourth team performed tabletop exercises. In addition, the inspector discussed with licensee representatives the current strategy. Target sets established by the licensee for the 1995 Operational Safeguards Response Evaluation (OSRE) remained current.

c. Conclusion

The licensee had in place a sound strategy that was capable of protecting vital equipment from acts intended to cause a significant release of radioactivity.

S8 Miscellaneous Security and Safeguards Issues

S8.1 Actions on Previous Inspection Findings (92904)

(Open) IFI 50-348, 364/97-02-01: Failure to Provide Locks of Substantial Strength to Prevent Tampering

The licensee had changed FNP-0-SP-10, "Patrol Procedures," Rev. 16, to require the motor patrol to physically check the locks every four hours, once per shift. In addition, the locks selected by the licensee were susceptible to damage by hand tools, creating a possible vulnerability. The licensee had purchased more substantial locks. However, the evaluation of the lock covers was still underway. This IFI remains open pending the completion of the licensee's evaluation.

(Closed) IFI 50-348, 364/97-13-01: Questionable Planned Biometrics Implementation.

The licensee was installing and testing the use of biometrics for access control in the protected areas. The licensee had determined that the Service Water Intake Structure (SWIS), a separate protected area, would be controlled by portable biometric units. Procedures were in place to process employees from one protected area to the other. This IFI is closed.

S8.2 Protection of Safeguards Information

a. Inspection Scope (81810)

An evaluation of the licensee's program to protect Safeguards Information (SGI) under the provision of 10 CFR 73.21 was conducted.

b. Observations and Findings

The inspector reviewed and evaluated FNP-0-AP-72, "Protection of Safeguards Information," and determined that all components of 10 CFR 73.21 were incorporated. The licensee currently is storing Safeguards Information at various locations. The inspector toured all areas and randomly checked General Services Administration (GSA) approved safes to ensure that they were locked. In addition, in Document Control, the inspector selected non-Safeguards approved containers and selected files to ensure that SGI was not being stored at these locations. All SGI was appropriately stored.

Through discussion with licensee representatives, the inspector determined that SGI was logged, transported, and given to only those individuals with fingerprints on file and with a need to know.

c. Conclusions

Safeguards Information was appropriately handled and stored as specified in 10 CFR 73.21.

F2 Status of Fire Protection Facilities and Equipment

F2.1 (Open) URI 50-348, 364/98-01-10: Pre-Action Sprinkler System Failures (71750)

On May 19, 1998, a conference call was held between the resident staff, NRR, Plant Farley personnel, and Farley Project personnel in Birmingham to discuss the status of this issue. The licensee reported that an equipment problem root cause team had been assembled, including representation from the nuclear industry. The team had met at the vendor's site and reviewed the problem. No conclusion concerning the failure cause had been identified. However, the team continued its evaluation.

V. Management Meetings and Other Areas

X1 Review of Updated Final Safety Analysis Report (UFSAR) Commitments

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters. The inspectors identified that FSAR Section 5.2.2.3 stated, "Pressurizer pressure is sensed by fast response pressure transmitters with a time response of better than 0.2 seconds." This is faster than the acceptance criteria of 0.23 seconds used by the licensee for testing the pressurizer pressure transmitters. This is not safety-significant because the pressurizer pressure instruments currently installed have response times faster than 0.2 seconds. This was provided to the licensee for resolution.

X2 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on June 4 and June 25, 1998, after the end of the inspection period. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Badham, Supervisor, Safety Audit and Engineering Review (SAER)
P. Crone, Engineering Support (ES) Performance Supervisor
K. Dyer, Security Manager, Farley Nuclear Plant
T. Esteve, Planning & Control Supervisor
R. Fucich, ES Manager
S. Fulmer, Plant Training and Emergency Preparations Manager
S. Gates, Administration Manager
D. Grissette, Operations Manager
R. Hill, General Manager
D. Jones, Configuration Management Manager
W. Lee, Emergency Preparedness Coordinator (corporate office)

T. Livingston, Chemistry Superintendent
R. Martin, Maintenance Team Leader
M. Mitchell, HP Superintendent
C. Nesbitt, Assistant General Manager, Plant Support
W. Oldfield, Nuclear Operations Training Supervisor
L. Revels, Assistant Security Manager, FNP
M. Stinson, Assistant General Manager, Operations
R. Vanderbye, Emergency Planning Coordinator
G. Waymire, Technical Support Manager
G. Wilson, SNC Corporate Senior Engineer
R. Winkler, Engineering Group Supervisor, PMMS
B. Yance, Plant Modifications and Maintenance Support Manager

NRC

J. Zimmerman, NRR Project Manager

INSPECTION PROCEDURES USED

IP 37001: 10 CFR 50.59 Safety Evaluation Program
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 62707: Maintenance Observations
IP 71707: Plant Operations
IP 71711: Plant Startup from Refueling
IP 71750: Plant Support Activities
IP 73753: Inservice Inspection
IP 81700: Physical Security Program for Power Reactors
IP 81810: Control of Safeguards Information
IP 82701: Operational Status of the Emergency Preparedness Program
IP 83750: Occupational Radiation Exposure
IP 84750: Radioactive Waste Treatment, and Effluent and Environmental Monitoring
IP 86750: Solid Radioactive Waste Management and Transportation of Radioactive Materials
IP 92700: Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92903: Followup - Engineering
IP 92904: Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Type	Item Number	Description and Reference
Open		
IFI	50-348, 364/98-03-01	Inadequate Thread Engagement (Section M1.3).
IFI	50-364/98-03-03	Rod Control Fuse Failures (Section M1.4).
VIO	50-348, 364/98-03-04	Inadequate Safety Assessment for Mis-wired Hot Shutdown Panel MOVs (Section E8.1).
EEI	50-348, 364/98-03-05	HSDP Loss of Function Inadequate Safety Evaluation (Section E8.1).
VIO	50-348, 364/98-03-06	Inadequate Corrective Actions for MCR Ventilation System (Section E8.5).
VIO	50-348, 364/98-03-07	Failure to Promptly Terminate Security Access (Section S2.1).
Closed		
NCV	50-364/98-03-02	Failure to Report Manual Reactor Trip in a Timely Manner (Section M1.4).
LER	50-348, 364/97-10-01	Motor Operated Valve Local - Remote Control Circuit Wiring Discrepancies (Section E8.1).
URI	50-348, 364/98-01-04,	Inadequate Safety Assessment for Mis-wired Hot Shutdown Panel MOVs (Section E8.1).
VIO	50-348, 364/97-11-04	Failure to Implement a Test Program for Service Testing of the TDAFW Battery (Section E8.2).
URI	50-348, 364/98-01-05	Failure to Track and Correct Conditions Adverse to Quality (Section E8.5).
IFI	50-348, 364/98-01-06	Control Room Ventilation Testing (Section E8.5).
IFI	50-348, 364/98-01-07	Review Licensee Actions to Improve Radioactive Material Container Label Effectiveness (Section R8.1).

IFI	50-348, 364/96-14-01	Exercise Weakness--Significant Emergency Information Was Not Communicated to the Appropriate Emergency Manager in a Timely Manner (Section P8).
IFI	50-348, 364/97-13-01	Questionable Planned Biometrics Implementation (Section S8.3).

Discussed

VIO	50-348, 364/97-11-03	TDAFW Battery Installation and Check Valve Test Deficiencies (Section E8.3).
URI	50-348, 364/98-01-10	Pre-Action Sprinkler System Failures (Section F2.1).
IFI	50-348, 364/97-02-01	Failure to Provide Locks of Substantial Strength to Prevent Tampering (Section S8.1).