UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

July 30, 2007

Southern Nuclear Operating Company, Inc.
ATTN: Mr. J. Randy Johnson
Vice President- Farley
Joseph M. Farley Nuclear Plant
7388 North State Highway 95
Columbia, AL 36319

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000348/2007003 AND 05000364/2007003

Dear Mr. Johnson:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Joseph M. Farley Nuclear Plant, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 12, 2007, with yourself and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, no findings of significance were identified by the NRC. However, licensee identified violations which were determined to be of very low safety significance are listed in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance of the violations and because they are entered into your corrective action program. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Farley Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

Sincerely,

/RA/

Scott M. Shaeffer, Chief
Reactor Projects Branch 2
Division of Reactor Projects

Docket Nos.: 50-348 and 50-364
License Nos.: NPF-2 and NPF-8

Enclosure: Inspection Report 05000348/2007003 and
05000364/2007003
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to J. Randy Johnson from Scott M. Shaeffer dated July 30, 2007

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000348/2007003 AND 05000364/2007003

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REGION II

Docket Nos.:	50-348, 50-364, 72-42
License Nos.:	NPF-2, NPF-8
Report Nos.:	05000348/2007003 and 05000364/2007003
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Joseph M. Farley Nuclear Plant
Location:	Columbia, AL 36319
Dates:	April 1 - June 30, 2007
Inspectors:	E. Crowe, Senior Resident Inspector J. Baptist, Resident Inspector S. Sandal, Resident Inspector W. Fowler, Reactor Inspector (Section 4OA5) W. Lewis, Reactor Inspector (Section 4OA5) B. Miller, Reactor Inspector (Section 1R08) R. Chou, Reactor Inspector (Section 1R08)
Accompanying Personnel:	M. Coursey
Approved by:	Scott M. Shaeffer, Chief Reactor Projects Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000348/2007003 and 05000364/2007003; 04/01/2007-06/30/2007; Joseph M. Farley Nuclear Plant, Units 1 & 2, Routine Integrated Report.

The report covered a three-month period of inspection by the resident inspectors and four reactor inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, Reactor Oversight Process.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program (CAP). These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at full Rated Thermal Power (RTP) and operated at full power for the duration of the inspection report period.

Unit 2 began the inspection period at full RTP until April 7, 2007, when the unit was shut down to begin a refueling outage. The unit was synchronized to the electrical grid on May 23, 2007 and achieved full RTP on May 29, 2007. The unit operated at full power for the remainder of the inspection report period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns. The inspectors performed partial walk-downs of the following three systems to verify the operability or redundant or diverse trains and components when safety equipment was inoperable. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The walk-downs were performed using the criteria in licensee procedures FNP-0-AP-16, Conduct of Operations - Operations Group, and FNP-0-SOP-0, General Instructions to Operations Personnel. The walk-downs included reviewing the Updated Final Safety Analysis Report (UFSAR), plant procedures and drawings, checks of control room and plant valves, switches, components, electrical power line-ups, support equipment, and instrumentation. Documents reviewed are listed in the Attachment.

- Unit 2 'B' Train Spent Fuel Pool (SFP) Cooling with full core offload during 'A' Train electrical outage
- Unit 1 'B' Component Cooling Water (CCW) during planned maintenance to 'A' CCW heat exchanger
- Unit 1 'A' Train Residual Heat Removal (RHR) system during planned maintenance to 'B' RHR pump containment sump suction isolation valve MOV8811B

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Fire Area Tours. The inspectors conducted a tour of the six fire areas listed below to assess the material condition and operation status of fire protection features. The inspectors verified that combustibles and ignition sources, were controlled in accordance with the licensee's administrative procedures; fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition, and that compensatory measure for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the requirements of licensee procedures FNP-0-AP-36, Fire Surveillance and Inspection; FNP-0-AP-38, Use of Open Flame; FNP-0-AP-39, Fire Patrols and Watches; and the associated Fire Zone Data sheets. Documents reviewed are listed in the Attachment.

- Unit 2 containment, Fire Zone 55
- Unit 2 reactor trip switchgear room, Fire Zone 20
- Unit 2 electrical penetration room, Fire Zone 34
- Unit 1 emergency diesel generator (EDG) 1-2A room, Fire Zone 61
- Unit 1 A RHR pump room, Fire Zone 1
- Unit 1 cable spreading room, Fire Zone 40

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

External Flooding. The inspectors reviewed the design, material condition, and procedures for coping with the design basis probable maximum flood. The inspectors reviewed the flooding sections of the UFSAR to determine the barriers required to mitigate the flood. The inspectors reviewed piping layout drawings and walked down the manholes for underground piping to ensure that the service water (SW) system would remain available following the probable maximum flood. The inspectors also reviewed the licensee’s analysis for the use of cable insulation degradation due to moisture in the manholes. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood. The flooding AOP also included provisions for installing spool pieces in different sections of piping throughout the plant. The inspectors walked down the fuel pool cooling heat exchangers, the component cooling heat exchangers, and associated SW piping to verify that these pieces were properly staged.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems ISI

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries for Unit 2. The inspectors observed and reviewed a sample of American Society of Mechanical Engineers (ASME) Section XI and Risk Informed ISI required examinations. The inspectors conducted a review of the following nondestructive examination (NDE) activities to evaluate compliance with Technical Specifications (TS), ASME Section XI and ASME Section V requirements - 1989 Edition, and to verify that indications and defects were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI, IWB/IWC-3000 acceptance standards.

Ultrasonic Testing (UT)
- 31" F2 APR1-4200-26RDM-RI, Steam Generator (SG) B Cold Leg Nozzle to Safe End

Visual Testing (VT-3)
- F2 APR1-4204-2SI-R140, Hydraulic Snubber on Safety Injection Discharge Piping
- F2 APR1-4204-2SI-R141, Two Directional Restraint on Safety Injection Discharge Piping

The inspectors reviewed the following UT examination records in addition to the records for the above observed examinations:

- 15" F2-APR1-4500-7DM-RI, Pressurizer Surge Nozzle to Safe End
- 6" F2 APR1-4205-49DM-AUG-UT, Pressurizer Spray Nozzle to Safe End

The inspectors reviewed the following examination record that contained recordable indications and verified that the indications were appropriately dispositioned in accordance with the ASME Code:

- 15" F2-APR1-4500-7DM-RI, Pressurizer Surge Nozzle to Safe End

Examiner qualifications, equipment certifications, and applicable NDE procedures for the above ISI examination activities were reviewed and compared to the applicable requirements of ASME Sections V and XI.

The inspectors also reviewed weld data sheets, welding procedures, and NDE results for the following weld performed since the last Unit 2 refueling outage.

- Q2E21P002A, 2A Charging Pump Discharge to Pipe, ASME Code Class 2

b. Findings

No findings of significance were identified.

.2 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors reviewed the licensee's BACC program (BACCP) to ensure compliance with commitments made in response to NRC Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and Bulletin 2002-01. "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The inspectors conducted an on-site record review and an independent walk-down of the containment building, which is not normally accessible during at-power operations, to evaluate licensee compliance with their program procedures and applicable industry guidance. In particular, the inspectors verified that the licensee's visual examinations focused on locations where boric acid leaks could cause degradation of safety significant components and that degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed the following three engineering evaluations completed for boric acid found on reactor coolant system piping and components to verify that leak evaluations were being properly completed in accordance with program and procedure requirements. The inspectors also reviewed licensee corrective action documents initiated for evidence of boric acid leakage to confirm that they were consistent with requirements of Section XI of the ASME Code, 10 CFR 50 Appendix B Criterion XVI, and licensee BACCP procedures.

- Q2G31H0001B, 2B SFP Heat Exchanger, Tube Sheet to Bell End Gasket Joint, Leak number 2807
- Q2B13V0077B, Pipe Cap Leak onto 2A Reactor Coolant Drain Tank (RCDT) Motor, Leak number 2840
- Q2E21V0407B, Pipe Cap Leak onto 2B RHR Piping, Leak number 2841

b. Findings

No findings of significance were identified.

.3 Steam Generator (SG) Tube ISI

a. Inspection Scope

The inspectors reviewed activities, plans, condition monitoring and operational assessments, the pre-outage degradation assessment, and procedures for the inspection and evaluation of the steam generator Inconel Alloy 690TT tubing for Unit 2 SGs A, B, and C, to determine if the activities were being conducted in accordance with TS and applicable industry standards. Data gathering, analysis, and evaluation activities were reviewed. The inspectors reviewed data results for tubes at SG A - R29C35, R35C35, R30C35, R41C26, R42C27, and R42C28; SG B - R08C35, R09C35, R09C36, R06C36, and R24C35; and SG C - R29C35, R29C19, and R30C19 to verify the adequacy of the licensee's primary, secondary, and resolution analyses. The inspectors reviewed video tapes which the licensee recorded during the visual inspection of the feedrings and photos which were taken during the inspection of the upper internals, including the steam drum areas to determine component conditions and detect foreign objects or loose parts inside the feedrings. The inspectors examined the foreign objects retrieved from the feedrings. The inspectors reviewed three condition reports (CRs) which were entered in the CAP for the evaluation and disposition of unusual conditions identified. The inspectors also reviewed data operators and analysts' certifications and qualifications, including medical exams.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

Quarterly Resident Review. On June 7, 2007, the inspectors observed portions of the licensed operator training and testing program to verify implementation of procedures FNP-0-AP-45, Farley Nuclear Plant Training Program; FNP-0-TCP-17.6, Simulator Training Evaluation/Documentation; and FNP-0-TCP-17.3, Licensed Retraining Program Administration (Classroom). The inspectors observed simulator scenario 2006/2008-S4-S604 conducted in the licensee's simulator for a loss of switchyard 500KV transmission line, 1C charging pump trip, failure of chemical volume and control system letdown pressure control, and a loss-of-coolant accident (LOCA) with associated ALERT emergency declaration. The inspectors observed high risk operator actions, overall performance, self-critiques, training feedback, and management oversight to verify operator performance was evaluated against the performance standards of the licensee's scenario. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two samples listed below for items such as: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); and (8) appropriateness of performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as (a)(1). In addition, the inspectors specifically reviewed events where ineffective equipment maintenance has resulted in invalid automatic actuations of Engineered Safeguards Systems affecting the operating units. Documents reviewed are listed in the Attachment.

- Unit 1 Auxiliary Feedwater (AFW) System
- Unit 2 AFW System

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following four activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee's risk assessment tool and risk categories in accordance with the requirements in licensee procedures FNP-0-ACP-52.3, Mode 1, 2, & 3 Risk Assessment; NMP-GM-006, Work Management; and FNP-0-AP-16, Conduct of Operations - Operations Group.

- April 12, Unit 2 ORANGE Risk Condition at Reactor Vessel Midloop Water Level
- April 16, Unit 2 YELLOW Risk Condition from 'B' Train Power Outage effect on Core Cooling and SFP Reliability
- May 3, Unit 2 ORANGE Risk Condition at Reactor Vessel Midloop Water Level
- May 13, Unit 1 YELLOW Risk Condition from 1A RHR Containment Sump Suction Isolation Valve maintenance

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following five operability evaluations to verify they met the requirements of licensee procedures FNP-0-AP-16, Conduct of Operations - Operations Group and FNP-0-ACP-9.2, Operability Determination (OD) for technical adequacy, consideration of degraded conditions, and identification of compensatory measures. The inspectors reviewed the evaluations against the design bases, as stated in the UFSAR and FSDs to verify system operability was not affected.

- CR 2007103104, U2 Solid State Protection System (SSPS), FNP-2-STP-46.0, Verification of Feedwater Isolation From Slave Relays and Isolation Valves, Card Failure
- CR 2007103055/2007103056, U2 FNP-2-STP-45.7, Main Steam Isolation Valve (MSIV) and Bypass valves In-service Test, failure at shutdown
- CR 2007103780 and 2007102101 1C EDG middle of cycle (MOC) switch failure
- CR 2007103548/2007103561, 1A RHR pump discharge line snubber leak
- 2A Feedwater Regulating Valve failure to achieve feedwater flow rate for 100% rated thermal power

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the criteria contained in licensee procedures FNP-0-PMT-0.0, Post-Maintenance Test Program, to verify post-maintenance test procedures and test activities for the following four systems/components were adequate to verify system operability and functional capability. The inspectors also witnessed the test or reviewed the test data to verify that test results adequately demonstrated restoration of the affected safety function(s).

- FNP-2-STP-80.15, DG 'B' Train loss of offsite power (LOSP) Sequencer B2J Load Shedding Circuit 2C EDG Station Blackout Start Test, following EDG MOC switch adjustment
- FNP-2-STP-40.0, Safety Injection With LOSP Test
- FNP-2-STP-45.7, MSIV and Bypass valves In-service Test, Cold Shutdown Valve Exercise Test Following MSIV Internal Components Replacement
- FNP-2-STP-627.0, Local Leak Rate Testing of Containment Penetrations for containment penetration 11 following failure of the SW From reactor coolant pump (RCP) Motor Air Cooler Isolation Valve MOV 3131 failing its as left test

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

Refueling Activities. The inspectors reviewed the following activities related to the Unit 2 refueling outage for conformance to licensee procedure FNP-0-UOP-4.0, General Outage Operations Guideline, and FNP-1-UOP-4.1, Controlling Procedure for Refueling. Surveillance tests were reviewed to verify results were within the T.S. required specification. Shutdown risk, management oversight, procedural compliance, and operator awareness were evaluated for each of the following activities. Documents reviewed are listed in the Attachment.

- Outage Risk Assessment
- Cooldown
- Core offload and reload
- Reactor coolant instrumentation
- Electrical system alignments and bus outages
- Reactor vessel disassembly and assembly activities
- Outage-related surveillance tests
- Containment Closure
- Low Power Physics Testing and Startup Activities
- Clearance Activities
- Decay Heat Removal and SFP Cooling
- Containment heavy load lifts

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed surveillance test procedures and either witnessed the test or reviewed test records for the following five surveillance tests to determine if the tests adequately demonstrated equipment operability and met the TS requirements. The inspectors reviewed the activities to assess for preconditioning of equipment, procedure adherence, and valve alignment following completion of the surveillance. The inspectors reviewed licensee procedures FNP-0-AP-24, Test Control; FNP-0-M-050, Master List of Surveillance Requirements; and FNP-0-AP-16, Conduct of Operations - Operations Group; and attended selected briefings to determine if procedure requirements were met. Documents reviewed are listed in the Attachment.

Surveillance Tests
- FNP-2-STP-46.0, Verification of Feedwater Isolation From Slave Relays and Isolation Valves In-service Test
- FNP-2-STP-40.1, B2F Sequencer Operability and Load Shedding Circuit Test

In-Service Test (IST)
- FNP-2-STP-45.7, MSIV and Bypass valves In-service Test.

Containment Isolation Valves

- FNP-2-STP-627, Q2E11V025A Local Leak Rate Testing of Containment Penetrations, Penetration 11
- FNP-2-STP-627, Q2E11V026A, Local Leak Rate Testing of Containment Penetrations, Penetration 11

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification (TM) and associated 10CFR50.59 screening criteria against the system design bases information and documentation and the licensee's TM procedure FNP-0-AP-8, Design Modification Control. The inspectors reviewed implementation, configuration control, post-installation test activities, drawing and procedure updates, and operator awareness for this TM. Documents reviewed are listed in the Attachment.

- CR 2007102897, Unit 1 Containment Purge Exhaust Fan

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee data for the PIs listed below to verify the accuracy of the PI data reported during the period listed. Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Rev. 4, was used to verify the basis in reporting for each data element.

Mitigating Systems Cornerstone
- Mitigating System Performance Index, Heat Removal Systems

The inspectors reviewed samples of raw PI data, Licensee Event Reports (LERs), and Monthly Operating Reports for the period covering March 1, 2006 through April 1, 2007.

The data reviewed from the LERs and Monthly Operating Reports was compared to graphical representations from the most recent PI report. The inspectors also examined a sampling of operations logs and procedures to verify that the PI data was appropriately captured for inclusion into the PI report as well as ensuring that the individual PIs were calculated correctly.

Barrier Integrity Cornerstone
- RCS Activity
- RCS Leakage

The inspectors reviewed raw PI data for the period of March 1, 2006, through April 1, 2007, consisting of daily chemistry analysis and daily leak rate logs. The inspectors reviewed the recent PI report to verify the data was accurately reflected in the report.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Daily Review

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing daily hard copy summaries of CRs and by reviewing the licensee's electronic CR database.

.2 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors review focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1. The review also included issues documented outside the normal CAP in system health reports, corrective maintenance WOs, component status reports, and MR assessments. The inspectors review nominally considered the six-month period of January through June 2007, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results contained in the licensee's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. The inspectors reviewed historical diesel generator (DG) CRs for small diameter line failures where the cause was attributed to vibration related fretting or fatigue failure. The review did not indicate any adverse trends with respect to DG small line failures induced by vibration related fretting or fatigue. Specific documents reviewed are listed in the attachment.

b. Assessment and Observations

The inspectors noted the following trends:

- The inspectors noticed an increasing negative trend in the area of procedure use and adherence issues. The inspectors noted the individual issues were generally spread across the general site population with the more significant issues related to the Operations Department. The licensee has generated CRs for each of the issues occurring during the first and second quarters of 2007. The licensee had previously identified this negative trend in the third quarter of 2006 as an emerging trend.

- The inspectors noticed the continuance of a negative trend in the area of 4160 circuit breaker failures. The licensee first identified the trend as an adverse trend in April, 2006. The licensee generated a long range plan to replace the aging Allis-Chambers breakers with Cutler-Hammer breakers. The rate of replacement was increased following failures during the fourth quarter of 2006. These failures have not affected more than one component/train at a time; therefore, the risk impact has been generally low. The licensee has replaced the breakers in the "H" and "J" buses which supply the non-safety related River Water Pumps and the "K" and "L" buses which supply the safety related SW Pumps. The licensee is currently experiencing fit-up issues with this modified mechanical device. The licensee's engineering staff is working with the vendor to resolve the issue. The licensee is replacing the remainder of the safety related breakers at the next opportunity. The NRC is conducting a follow up inspection in this area in accordance with Inspection Procedure 95001. This inspection will be documented in Inspection Report 2007-008.

.3 Work-Around Annual Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a detailed review of the work-around lists for Unit 1 and 2 that were in effect on June 15, 2007. The inspectors reviewed the proposed corrective actions and schedule for each item on the work-around list. The inspectors reviewed the compensatory actions and cumulative effects on plant operation. The inspectors verified each item was being dispositioned in accordance with plant procedure ACP-17, Operator Work-Around.

b. Findings and Observations

No findings of significance were identified. The inspectors found that operator work-arounds were being identified at an appropriate threshold. Items were entered into the CAP or actions taken were appropriate.

4OA3 Event Followup

.1 Unit 2 Overpower event

a. Inspection Scope

On June 11, an unexpected and brief power increase to 100.89% occurred due the inadvertent closure of an extraction steam valve for 6A feedwater heater. The inspectors discussed the event with operations, engineering, and licensee management personnel. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected.

b. Findings

No findings of significance were identified.

.2 Unit 2 Return to Mode 5 Following Obtaining Normal Operating Pressure and Temperature at End of Refueling Outage

a. Inspection Scope

On May 12, the licensee decided to reduce reactor coolant system pressure and temperature to inspect the seals of the A Reactor Coolant Pump due to reactor coolant pump seal leakoff rates that were less than normal. The inspectors discussed the unplanned mode changes with operations, engineering, and licensee management personnel. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected. The inspectors reviewed the additional risk to the plant for this evolution.

b. Findings

No findings of significance were identified.

.3 Unit 1 Main Steam Line Leak

a. Inspection Scope

On April 5, a steam leak developed on the transition joint for pressure transmitter PT-476 (Unit 1 steam generator 1A) downstream of the root isolation valve. The inspectors discussed the proposed power reduction to approximately 25% for valve repairs with operations and licensee management personnel. The licensee performed further evaluations and decided to remain at power, make additional attempts to close the root isolation valve to isolate the leak, and re-route instrument tubing to pressure transmitter PT-476. The inspectors reviewed the above decision with operations, engineering, and licensee management.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 (Discussed) Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-002) - Unit 2

a. Inspection Scope

The inspectors reviewed the documents listed in the Attachment to verify implementation of the licensee's commitments documented in their September 1, 2005, response to Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors. The commitments included permanent modifications, program and procedure changes and various analyses. Permanent modifications included installation of new containment sump strainers and increasing internal clearances for the High Head Safety Injection (HHSI) branch line throttle valves. Program and procedural changes included removal of refueling cavity drain covers, containment sump screen inspection, and enhancements for control of tags and labels in containment. Various analysis included in the GL 2004-002 response were debris generation, debris transport, Net Positive Suction Head (NPSH) calculations, screen testing, chemical effects, and downstream effects. Licensing basis documents were reviewed to verify updates associated with GL 2004-002 were identified and appropriate changes made. The inspectors conducted a walkdown to verify key sump screen features were appropriately translated into the final design and that screen installation was properly implemented. Plant modifications associated with installation of the sump screens were reviewed to verify these modifications were properly approved and did not impact plant safety.

b. Findings and Observations

No findings of significance were identified. The TI will remain open pending completion of the following GL 2004-02 commitments:

- Finalized NPSH calculations, Debris Generation, Debris Transport, and Chemical Effects analysis. Revisions are currently being implemented associated with containment coatings.

- Modification of the HHSI branch line throttle valves to meet minimum clearances as identified by the downstream effects analysis. This will require an NRC extension from the GL requirements if not implemented prior to December 31, 2007.

4OA6 Meetings, Including Exit

On July 12, 2007, the inspectors presented the inspection results to Mr. Randy Johnson and the other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information provided by the licensee was returned to the licensee at the completion of the inspection.

4OA7 Licensed-Identified Violations

The following findings of very low safety significance (Green) were identified by the licensee and are a violation of NRC requirements which meet the criteria of Section IV of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System, required two RHR suction relief valves, with setpoints less than or equal to 450 psig, be in service when the temperature of one or more RCS cold legs is less than or equal to 325°F. Contrary to this requirement, on May 11, 2007, the licensee allowed RCS temperature to reach 300°F for approximately 32 minutes with two RHR suction relief valves isolated. This was identified in the licensee's CAP as CR 2007104856. This finding is of very low safety significance because the valves were isolated for a short period of time and an overpressure condition did not exist.

- TS 5.4.1 requires written procedures to be established, implemented, and maintained covering the activities of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A of Regulatory Guide 1.33 stated, in part, that plant startup be covered by written procedures. Station Procedure FNP-2-UOP-1.1, Startup of Unit From Cold Shutdown to Hot Standby, step 3.17, required that primary-to-secondary steam generator differential pressure not exceed 1800 psid. Contrary to this requirement, on May 18, 2007, the licensee allowed primary-to-secondary steam generator differential pressure to increase to 1830 psid for approximately 32 minutes. This event is documented in the licensee's CAP as CR 2007105099. This finding is of very low safety significance because the differential pressure exceed the limit by a small amount and existed for a short period.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
W.L. Bargeron, Assistant General Manager - Operations
W. R. Bayne, Performance Analysis Supervisor
S. H. Chestnut, Engineering Support Manager
P. Harlos, Health Physics Manager
L. Hogg, Security Manager
J. Horn, Training and Emergency Preparedness Manager
J. Jerkins, Performance Analysis Engineer
J. Johnson, Plant Vice President
T. Livingston, Chemistry Manager
B. Moore, Maintenance Manager
W. Oldfield, Quality Assurance Supervisor
J. Swartzwelder, Work Control Superintendent
R. Vanderbye, Emergency Preparedness Coordinator
R. Wells, Operations Manager
T. Youngblood, Assistant General Manager - Plant Support

NRC personnel
S. Shaeffer, Division of Reactor Projects, Branch Chief

LIST OF ITEMS DISCUSSED

Discussed

2515/166 (Unit 2)	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-002)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment
Technical Specifications 3.5.2, 3.7.7, and 3.7.13
P&ID Drawing 178002 Sheet 1, 178002 Sheet 3,178038 Sheet 1, 178038 Sheet 2, 178041, 205043, 205002, 205038, 205041
UFSAR Section 5.5.7, Residual Heat Removal System
UFSAR Section 9.1.3, Spent Fuel Pool Cooling and Cleanup System
UFSAR Section 9.2.2, Cooling System for Reactor Auxiliaries
FNP-1-SOP-7.0, Residual Heat Removal System, Revision 73
FNP-1-SOP-7.0A, Residual Heat Removal System, Revision 6

Section 1R05: Fire Protection
Plant Drawings
A-508650, Sheet 6 Revision 1
A-508650, Sheet 30 Revision 11
A-508650, Sheet 31 Revision 12
A-508651, Sheet 6 Revision 3

A-509018, Sheet 18 Revision 11
A-509018, Sheet 26 Revision 1
A-509018, Sheet 32 Revision 7
A-509018, Sheet 33 Revision 8
A-509018, Sheet 46 Revision 2
A-509018, Sheet 47 Revision 2
A-509018, Sheet 48 Revision 1
A-509018, Sheet 49 Revision 1

Section 1R06: Flood Protection Measures
Farley Nuclear Plant Units 1 and 2 Individual Plant Examination Report In Reponse to Generic Letter 88-20
FNP-2-ARP-3.1, Annunciator Response Procedure for BOP Panel L, Revision 15
FNP-2-ARP-3.2, Annunciator Response Procedure for BOP Panel N, Revision 17

Section 1R08: Inservice Inspection (ISI) Activities
Procedures
Southern Nuclear Operating Company, NMP-ES-024-507, PDI Generic Procedure for the Ultrasonic Examination of Dissimilar Metal Pipe Welds (Appendix VIII), Ver 1.0
Southern Nuclear Operating Company, NMP-ES-019-001, Boric Acid Corrosion Control Program Implementation, Ver 1.0
Farley Nuclear Plant, FNP-0-M-101, Boric Acid Corrosion Control Program, Ver 11
Farley Nuclear Plant, FNP-0-EPT-4496, Corrosion Assessment, Ver 2.0
Southern Nuclear Operating Company, NMP-ES-024-100, Procedure For Qualification and Certification of Nondestructive Examination Personnel, Version 2.0
Farley Nuclear Plant, FNP-0-AP-31, Quality Control Measures, Version 17.0
Farley Nuclear Plant, FNP-0-PMP-514, Visual Inspection of Welds, Revision 8
Westinghouse MRS-SSP-1169-ALA/APR, Rev. 3, Farley Units 1 & 2 In-Service Steam Generator Eddy Current Analysis Guidelines
Westinghouse MRS-SSP-1051-ALA/APR, Rev. 4, Upper Internals Inspection - Westinghouse Model 54F S/Gs
Westinghouse MRS 2.4.2 GEN-35, Rev. 12, Eddy Current Inspection Of Preservice and Inservice Heat Exchanger Tubing
Westinghouse MRS-TRC-1797, Use of Appendix H Qualified techniques at Farley Unit #2R18 Spring 2007 S/G Inspection

Section 1R11: Licensed Operator Requalification
FNP-0-AP-45, Farley Nuclear Plant Training Program, Revision 22
FNP-0-TCP-17.3, Licensed Retraining Program Administration (Classroom), Revision 32
FNP-0-TCP-17.6, Simulator Training Evaluation/Documentation, Revision 16
FNP-2-EEP-0.0, Reactor Trip or Safety Injection, Revision 28
FNP-2-EEP-1.0, Loss of Reactor or Secondary Coolant, Revision 25
FNP-2-AOP-1.0, RCS Leakage, Revision 14
FNP-2-AOP-16.0, CVCS Malfunction, Revision 10
FNP-2-AOP-17.0, Rapid Load Reduction, Revision 15
Scenario 2006/2008-S4-S604

Section 1R12: Maintenance Effectiveness
Condition Reports: 2005112444, 2006102769, 2006105351, 2006107828, 2006108737, 2007101332, 2007102460
FNP-0-M-87, Maintenance Rule Scoping Manual, Revision 16
FNP-0-M-89, FNP Maintenance Rule Site Implementation Manual, Revision 12
Functional System Description A-181010, Auxiliary Feedwater
NMP-ES-021, Structural Monitoring Program for the Maintenance Rule, Revision 2

Section 1R19: Post Maintenance Testing
Condition Reports: 2007105147; 2007103277; 2007103362; 2007103804; 2007103832; 2007104000; 2007104001; 2007104042; 2007104367; 2007104372
FNP-0-GMP-27.0, General Maintenance Procedure, Revision 26.0
FNP-2-STP-40.0, Safety Injection With Loss of Off-Site Power Test, Revision 46
FNP-2-STP-627.0, Local Leak Rate Testing of Containment Penetrations, Revision 43
Work Orders: 2060530201; 2060968102; 2070827101; 2070827102

Section 1R20: Refueling and Other Outage Activities
FNP-0-UOP-4.0, General Outage Operations Guidance
FNP-0-AP-52, Equipment Status Control and Maintenance Authorization
FNP-2-UOP-4.1, Refueling Outage Operation
FNP-0-AP-94, Outage Nuclear Safety
FNP-2-UOP-4.3, Mid-Loop Operations
FNP-0-ACP-47.3, Outage Preparation
FNP-2-STP-35.0, Reactor Coolant System Pressure and Temperature/Pressurizer Temperature Limits Verification
FNP-2-UOP-2.1, Shutdown of Unit From Minimum Load to Hot Standby
FNP-2-UOP-2.2, Shutdown of Unit From Hot Standby to Cold Shutdown
FNP-2-SOP-1.6, Draining th Reactor Coolant System
FNP-2-SOP-1.3, Reactor Coolant System Filling and Venting-Vacuum Method
FNP-2-STP-18.4, Ctmt Mid-Loop and/or Refueling Integrity Verification and Ctmt Closure
FNP-2-IMP-201.45, Refueling Reactor Coolant System Level Calibration Q1B21FT0416
FNP-2-STP-35.1, Unit Startup Technical Specification Verification
FNP-0-ETP-3643, Verification of Rod Control System Availability
FNP-2-STP-101, Zero Power Reactor Physics Testing
FNP-2-STP-29.6, Calculation of Estimated Critical Condition
FNP-2-MP-11.4, Reactor Polar Crane - Operating and Safe Load Path Instructions
NMP-MA-007-001, SNC Rigging and Lifting Program Planning and Evaluation
NMP-MA-007-009, SNC Rigging and Lifting Program Plant Farley Specifics
Southern Nuclear Design Calculation 036.B, Analysis of Containment Operating Deck for Heavy Load Drops - NUREG 0612
Work Order: 2050929801

Section 1R22: Surveillance Testing
FNP-0-AP-24, Testing Control, Revision 8
FNP-0-AP-16, Conduct of Operations - Operations Group, Revision 42
FNP-0-M-50, Master List of Surveillance Requirements, Revision 21

FNP-1-SOP-17, Main and Reheat Steam, Revision 52
FNP-2-SOP-17, Main and Reheat Steam, Revision 42
FNP-1-STP-9, RCS Leakage Test, Revision 39
FNP-1-STP-15, Containment Air Lock Door Seal Operability Test, Revision 31
FNP-1-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test (Tave>547°F), Revision 43
FNP-2-STP-4.3, 2C Charging Pump Quarterly Inservice Test. Revision 39
FNP-2-STP-15, Containment Air Lock Door Seal Operability Test, Revision 27
FNP-2-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test (Tave>547°F), Revision 45

Section 1R23: Temporary Plant Modifications
Condition Reports: 2007102897
FNP-0-AP-8, Design Modification Control, Revision 38
Work Orders: 1070696502 and 1070696503

Section 4OA1: Performance Indicator Verification
Procedures, Manuals, and Guidance Documents
FNP-0-AP-54, Preparation and Reporting of NRC Performance Indicator Data and NRC Operating Data, Ver. 8.0
FNP-2-STP 726, Plant Vent Stack Contingency Sampling Ver. 15

Records and Data Reviewed
Electronic dosimeter dose rate alarm logs, April 2006 - December 2006
LCO-2-005-379, R-29B Inoperable for STP-814, 11/29/05
Special Report No. 2005-001-00, Inoperable Radiation Monitor R-29B, 50-364
FNP Surveillance Test Review Sheet, Surveillance Test No. FNP-2-STP-726, Technical Requirements Manual Reference TR 13.3.4, action C.1 and Offsite Dose Calculation Manual References 3.1.1.3 action 37, 3.1.1.3 action 39, 12/21/06-

CAP Documents
2006103204, 2006105114, 2006108879

Section 4OA5: Other
Worksheet 2050912001C001, Summary of Supporting Modifications, Rev. 1.0
DCP 2050912001, Containment Sump Screen Mods, Rev. 1.0
FCR 2050912001-001, Mtl/Weld Change Incorporation & Guidance for Plugging Strainer Holes Exceeding 3/32"
FCR 2050912001-002, Mtl/Weld Change Incorporation & Minor Dimensional Corrections
FCR 2050912001-003, Configuration Changes to Support Field Conditions (Including Sump Level Transmitters Q2E11LT3594A and 359B)
FCR 2050912001-004, Configuration Changes to Support Field Conditions (Including Charging Pump Discharge Orifice Restriction for Q2E21P002A, B and C)
FCR 2050912001-005, Mtl Change Incorporation
FCR 2050912001-006, Weld Change Incorporation
FCR 2050912001-007, Configuration Changes to Support Field Conditions (Including

Relocation Point for Sump Level Transmitter Q2E11LT3594A)
FCR 2050912001-008, Addition of Hilti-bolts to Make Up for Reduced Embedment
FCR 2050912001-009, Reduced Hilti-bolting for Relocated TSP Baskets
FCR 2050912001-010, Reduced Anchor Bolt Setback from Edge of Concrete and Reduced Hilti-bolting to Support Field Conditions
FCR 2050912001-011, Mtl/Weld Change Incorporation
DOEJ-SC-2050912001-001, Justification for the Temporary Removal of Kicker from Support Cluster During Farley Unit 2 Outage (F2R18), Ver. 1.0
DOEJ-SC-2050912001-002, Evaluation of Minor Modifications to Stair No. 4 to Accommodate New Unit 2 Containment Sump Screen RHR "Bravo," Ver 1.0
DOEJ-SC-2050912001-003, Evaluation of Watertight Doors No 177 & 2177 in Plant Farley Unit 1 & 2 Containment, Ver 1.0
DOEJ-SC-2050912001-004, Evaluation of Conduit & Pull/Junction Box Support, Ver 1.0
DOEJ-SC-2050912001-005, Civil Evaluation of Water Level Orifice Assembly Mounting Detail Ver 1.0
DOEJ-SC-2050912001-006, Civil Review of General Electric's Anchor Bolt Design for Containment Sump Strainers and Pipe Anchors, Ver 1.0
CN-SEE-05-93, Farley Unit 1 & Unit 2 NPSH Calculation form the Containment Sump to the RHR Pumps (Westinghouse Calculation), Rev. 0
CN-SEE-05-99, Farley Unit 1 & Unit 2 NPSH Calculation form the Containment Sump to the Containment Spray Pumps (Westinghouse Calculation), Rev. 0
DOEJ-SM-FC053044801-001, Containment Overpressure for Sump Screen Flashing Evaluation, Ver 1.0
DOEJ-SM-2050912001-001, Evaluation of Orifice Thickness for CTMT Floor & Equipment Drain Orifice N2G21FO0001 & N2G21FO0002, Ver 1.0
2005-05680, GSI-191 Downstream Effects - Flow Clearances, Rev. 0
SM-2050912001-001, HHSI Throttle Valve Clearance and Orifice Sizing, Ver. 1.0
QP050103, Containment Sump Screen Purchase Order, 2/16/05
QP050103 Receipt Inspection, 4/9/07
Report No. 06-05, Data Report for Farley Sector Tests, 2/2006
NEL 96-0453, ECCS Throttle Valve Clogging, 12/30/1996
U-611412, Worksheet 2050912001M034, Design Specification - Containment Sump Passive Strainer/Screen, Ver. 1.0
U-611407, Worksheet 2050912001M029, General Arrangement - Containment Sump Passive Strainer/Screen, Ver. 1.0
U-611410, Worksheet 2050912001M032, Installation Specification - Containment Sump Passive Strainer/Screen, Ver. 1.0
U-611405, Worksheet 2050912001M027, Stress Analysis - Containment Sump Passive Strainer, Ver. 1.0

Procedures
FNP-0-AP-25.0, Equipment Identification and Labeling, Ver. 13.0
FNP-2-MP-1.0, Maintenance Refueling Procedure, Ver. 46.0
FNP-2-STP-34.0, Containment Inspection (General), Ver. 20.0
FNP-2-STP-34.2, Containment ECCS Sump Intake Inspection, Ver. 4.0
FNP-2-UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby, Ver. 75.0

Drawings
U-276739 A, 2" 1500# Y-Type Throttle Valve w/ Lock, Rev. A0
D-205200, Containment Floor & Equip Drains - Plan at El. 105'-6", Rev 10
D-206200, Floor Plan at El. 105'-6" - Concrete Containment, Rev 5