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

July 28, 2009

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

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000348/2009003 AND 05000364/2009003

Dear Mr. Johnson:

On June 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Joseph M. Farley Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on July 9, 2009, with 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 NRC reviewed selected procedures and records, observed activities, and interviewed personnel.

The report documents two NRC-identified findings and one self-revealing finding of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. Additionally, four licensee-identified violations (LIVs), which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCV) consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV, 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 addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Farley Nuclear Plant. The information you provide will be considered in accordance with the Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, 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 NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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 No.: 50-348, 50-364
License No.: NPF-2, NPF-8

Enclosure: Inspection Report 05000348/20009003 and 05000364/2009003
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 28, 2009

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000348/2009003 AND 05000364/2009003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	05000348, 05000364
License Nos.:	NPF-2, NPF-8
Report No.:	05000348/2009003 and 05000364/2009003
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Joseph M. Farley Nuclear Plant, Units 1 and 2
Location:	Columbia, AL
Dates:	April 01, 2009 through June 30, 2009
Inspectors:	E. Crowe, Senior Resident Inspector J. Hickey, Senior Resident Inspector, Hatch S. Sandal, Resident Inspector C. Rapp, Senior Project Engineer G. Kuzo, Senior Health Physicist (Section 2OS1) A. Nielsen, Health Physicist (Section 2OS1) B. Collins, Reactor Inspector (Section 1R08, 4OA5)
Approved by:	Scott M. Shaeffer, Chief Reactor Projects Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000348/2009-003 and 05000364/2009-003; 04/01/2009 – 06/30/2009; Joseph M. Farley Nuclear Plant, Units 1 and 2; Post Maintenance Testing, Identification and Resolution of Problems, Surveillance Testing

The report covered a three-month period of inspection by the resident inspectors, a visiting senior resident inspector, a reactor inspector, a senior health physicist, a health physicist and a senior project engineer. Two NRC identified findings and one self-revealing finding with very low safety significance (GREEN) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. An NRC identified NCV of 10 CFR 50, Appendix B, Criterion XI, Test Control, was identified for failure to establish adequate procedures to verify automatic closure of the Unit 1 and Unit 2 Steam Generator blowdown (SGBD) isolation valves. This finding has been entered into the licensee's CAP as condition report (CR) 2009107127.

  The failure to establish and implement adequate test procedures including acceptance criteria necessary to verify safety-related equipment is capable of performing its design function is a performance deficiency. This finding is more than minor because if left uncorrected, the condition could result in the failure to recognize safety function testing acceptance criteria specified by plant design had not been met, and would be a more significant safety concern. This finding was assessed using the Phase 1 screening worksheet of the SDP and determined to be of very low safety significance (Green), because it did not result in an actual loss of safety function of a single train for greater than the Technical Specification (TS) allowed outage time, and was not potentially risk-significant due to external events. This finding was assigned a cross-cutting aspect in the Problem Identification and Resolution (PI&R) area (P.1(d)) because the licensee had identified the need for additional testing of contacts related to SGBD isolation in March 2006, but no corrective actions were taken to address this issue in a timely matter commensurate with safety significance and complexity. (Section 1R19)

- Green. A self-revealing NCV of TS 5.4.1.a. for failure to implement FNP-1-STP- 33.8, Verification of Steam Dump Arming from Reactor Trip (P-4) Signal and P-4 Contact Verification, was identified when licensee personnel took actions not directed by the procedure. During performance of FNP-1-STP- 33.8, licensee personnel changed the established initial conditions by placing the Solid State Protection System (SSPS) into a different operating mode. This finding has been entered into the licensee's CAP as CR 2009104718.

  The failure to follow FNP-1-STP- 33.8 is a performance deficiency. This finding is more than minor because it was associated with the Mitigating Systems cornerstone attribute of Configuration Control and adversely affected a cornerstone objective in that failure to follow the procedure resulted in changing the initial conditions previously established. This finding was determined to be of very low safety significance because the procedure was successfully preformed prior to entering a mode that required the SSPS to be operable. This finding has a cross-cutting aspect of Work Control in the area of Human Performance (H.2(b)) in that the licensee did not keep personnel apprised that the SG LO-LO level reactor trip signal in SSPS would not be blocked using the "Normal Method." (Section 1R22)

- Green. An NRC identified NCV of 10 CFR 50.65 (b) was identified for failure to include the Auxiliary Building water-tight doors within the scope of the maintenance rule (MR) monitoring program. During routine plant inspections, the inspectors noticed degraded door seals (excessive hardening of the rubber seal with small chunks missing) and worn hinges on water tight doors located in the Auxiliary Building. The inspectors discovered one Unit 2 door in which the seal area had cracks completely across. The inspectors determined the sealing function assumed in the licensee internal flooding analysis for these doors was challenged. This finding has been entered into the licensee's CAP as CR 2009106669.

  The licensee's failure to include the water-tight doors within the scope of their MR program and the subsequent degraded condition of these doors is a performance deficiency. The performance deficiency is more than minor because it adversely affected the equipment performance attribute of the Mitigating System Cornerstone objective to ensure the availability, reliability, and capability of systems responding to initiating events to prevent undesirable consequences (i.e. core damage). This finding was assessed using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined the finding was of very low safety significance (Green) because although it did involve degradation of equipment or function specifically designed to mitigate a flooding event (i.e. flooding barriers), it did not result in a loss of safety function. No cross cutting aspects were identified. (Section 4OA2)

B. Licensee-identified Violations

Violations of very low safety significance, identified by the licensee, have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's CAP. These violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 started the report period at 90 percent Rated Thermal Power (RTP). The unit was shut down on April 4 to begin refueling outage (RFO) 22. The reactor was made critical on May 6 and achieved 100 percent RTP on May 13. The unit remained at or near 100 percent RTP until June 26 when power was reduced to 61 percent RTP for emergent repairs to the 1A main feedwater (FW) pump auto stop oil pressure switch. The unit was returned to 100 percent RTP on June 27 and remained at or near 100 percent RTP for the remainder of the inspection period.

Unit 2 started the report period at 12 percent RTP. The unit achieved 100 percent RTP on April 4. The unit remained at or near 100 percent RTP until June 27 when power was reduced to 61 percent RTP for planned repairs to the 2A main FW pump lubricating oil cooler. The unit returned to 100 percent RTP on June 29 and remained at or near 100 percent RTP for the remainder of the inspection period.

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

1R01 Adverse Weather Protection

a. Inspection Scope

Grid Reliability: The inspectors reviewed the licensee's station procedures to verify communication protocols exist between the transmission operator and the control room to promptly identify issues that could impact the offsite power system. The inspectors verified adequacy of these procedures to address measures to monitor and maintain availability and reliability of the offsite alternating current (AC) power system and the onsite alternate AC power system. The inspectors also reviewed the compensatory actions identified in station procedures to be performed when it is not possible to predict post-trip voltage at the site for current electrical grid conditions. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial Walk-Down: The inspectors performed partial walk-downs of the following four systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors attempted to identify discrepancies impacting the function of the system and therefore, potentially increasing risk. The walk-downs were performed using the criteria in licensee procedures NMP-OS-007, Conduct of Operations, 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, support equipment, and instrumentation. Documents reviewed are listed in the Attachment.

- Unit 1 A and B Residual Heat Removal (RHR)
- Unit 1 Spent Fuel Pool (SFP) Cooling System during complete reactor core off-load
- Unit 1 safety-related Electrical Distribution System during mid-loop refueling activities
- Unit 2 safety-related Electrical Distribution System during 2B Emergency Diesel Generator (EDG) 24 month overhaul

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Fire Protection Area Tours: The inspectors conducted a tour of the four fire areas listed below to assess material condition and operation status of the fire protection equipment. The inspectors verified combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures; fire detection and suppression equipment was available for use; passive fire barriers were maintained in good material condition, and compensatory measures 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 1, Containment, Fire Zone 55
- Unit 1, Component Cooling Water (CCW) Equipment Room, Fire Zone 6
- Unit 1, Electrical Penetration Room, Fire Zones 34 and 35
- Unit 1, 1B Motor Driven Auxiliary Feedwater Pump (MDAFWP) Room, Fire Zone 6

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

Non-Destructive Examination (NDE) Activities and Welding Activities: The inspector reviewed the implementation of the licensee's In-service Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary and risk-significant piping boundaries. The inspector's activities consisted of an on-site review of

NDE and welding activities to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2001 Edition with 2003 Addenda), and verification that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with requirements of the ASME Code, Section XI acceptance standards.

The inspector's review of NDE activities specifically covered examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, and calibration reports (as applicable) for the following examinations:

- Ultrasonic Testing (UT) examination of weld ALA1-4101-7, ASME Class 1, RHR System, 12-inch diameter elbow-to-pipe weld – Document Review
- UT examination of weld ALA1-4101-8, ASME Class 1, RHR System, 12-inch diameter valve-to-pipe weld – Document Review
- Penetrate Testing (PT) examination of welds ALA1-4101-RHR-R99 (W2), ASME Class 1, RHR System, 8-inch diameter welds – Direct Observation

The inspector's review of welding activities specifically covered the welding activities listed below in order to evaluate compliance with procedures and the ASME Code. The inspector reviewed the WOs, repair and replacement plans, weld data sheets, welding procedures, procedure qualification records, welder qualification records, and NDE reports.

- WO 1082199502 - Welding Package for installation of vent valve in response to Generic Letter (GL) 2008-01 (ASME Class 1).
- WO 1082199503 - Welding Package for installation of vent valve in response to GL 2008-01 (ASME Class 1).

<u>PWR Vessel Upper Head Penetration (VUHP) Inspection Activities</u>: Inspections during this outage consisted of visual examinations conducted above the reactor VUHP to identify potential boric acid leaks from pressure-retaining components. The inspector specifically reviewed examination procedures, personnel training and qualification records, reports for the visual inspection of pressure-retaining components above the head performed every outage and reviewed the licensee's calculations for effective degradation years (EDYs). The inspector verified compliance with the requirements contained in 10CFR50.55a(g)(6)(ii)(d) and Code Case N-729-1.

<u>Boric Acid Corrosion Control (BACC) Inspection Activities</u>: The inspector reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC GL 88-05, "BAC of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspector performed an on-site record review of procedures and the results of the licensee's containment walk-down inspections performed during the Unit 1 1R22 outage. The inspector also interviewed the BACC program owner and conducted a walk-down of the reactor building to evaluate compliance with the licensee's BACC program requirements and to verify degraded or non-conforming conditions, such as boric acid leaks identified during the containment

walk-down, were properly identified and corrected in accordance with the licensee's BACC and CAPs.

The inspector reviewed a sample of engineering evaluations completed for evidence of boric acid found on systems containing borated water to verify the minimum design code requiring section thickness had been maintained for the affected components. The inspector selected the following evaluations for review:

- CR 2009103927 – Boric Acid Leak Evaluation, RPVH Bolts 8 & 9.
- CR 2009103966 – Boric Acid Leak Evaluation, Leak on Pressurizer.
- CR 2009103971 – Boric Acid Leak Evaluation, Leak on Pressurizer.

Steam Generator (SG) Tube Inspection Activities: No SG tube inspection activities were conducted during this outage. The inspector discussed the status of foreign objects and associated foreign object search and recovery (FOSAR). The inspector reviewed the licensee's Secondary Side Integrity plan, and interviewed plant personnel to ensure compliance with Electric Power Research Institute (EPRI) SG Integrity Assessment Guidelines, Revision 2.

Identification and Resolution of Problems: The inspector performed a review of ISI-related problems, including welding, BACC, and SG inspections identified by the licensee and entered into the CAP as CRs. The inspector reviewed the CRs to confirm the licensee appropriately described the scope of the problem and initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspector performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspector are listed in the report attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

Resident Inspector Quarterly Review: On May 20, 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 Plan; FNP-0-TCP-17.6, Simulator Training Evaluation/Documentation and FNP-0-TCP-17.3, Licensed Operator Continuing Training Program Administration. The inspectors observed operations simulator scenario 09-S602, conducted in the licensee's simulator for a reactor trip with failure of auto turbine trip, the 1A battery input breaker trips, an unaffected unit Notification of Unusual Event (NOUE) declaration resulting from a loss of main control board annunciation, and a subsequent rapid load reduction of affected unit due to main condenser air leak resulting in a turbine trip. The inspectors observed high-risk operator actions, overall crew 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 Rule Effectiveness

a. Inspection Scope

The inspectors reviewed the following two activities for (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 NRC specifically reviewed events where ineffective equipment maintenance resulted in invalid automatic actuations of Engineered Safeguards Systems affecting the operating units. Documents reviewed are listed in the Attachment.

- CR 2008111986, Unit 2, Motor-Operated Valve (MOV) 3227B will not close from main control board
- CR 20081111182, Unit 2, Flow Control Valve (FCV) 3009C failed to fully open when demanded

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the following four activities to verify appropriate risk assessments were performed prior to taking equipment out of service (OOS) for maintenance. The inspectors verified 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 appropriate use of the licensee's risk assessment and risk categories in accordance with requirements in licensee procedures FNP-0-ACP-52.3, Mode 1, 2, & 3 Risk Assessment; FNP-0-UOP-4.0, General Outage Operations Guidance; NMP-GM-006, Work Management and NMP-OS-007, Conduct of Operations.

- Unit 1, April 7, Orange risk condition due to reduced RCS inventory with RCS level at the reactor vessel flange
- Unit 1, April 25, Orange risk condition due to reduced RCS inventory with RCS level at mid-loop for vacuum refill

- Unit 2, May 20, Yellow risk condition due to 2A MDAFW pump room cooler out of service coincident with missed surveillance for the HI-HI SG Level Trip Response Time Test
- Unit 2, June 22, Yellow risk condition due to 2B EDG 24 month overhaul and high voltage switchyard inspections

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 NMP-OS-007, Conduct of Operations and NMP-AD-012, ODs and Functionality Assessments. The scope of this inspection also included a review of the technical adequacy of the evaluations, the adequacy of compensatory measures, and the impact on continued plant operation.

- CR 2009103797, Unit 2, Instrument Air cross-connect to Emergency Air Compressor (Q2N11V016A) check valve back leakage
- CR 2009105601, Unit 1, Foreign material discovered in refueling cavity “sandboxes”
- CR 2009106789, Unit 2, 2C High Head Safety Injection (HHSI) pump breaker failed to open when demanded from main control board
- CR 2009107292, Unit 1/2, Valve encapsulations removed from both trains of RHR and Containment Spray
- CR 2009107457, Unit 1/2, EDG 1-2A high lubricating oil temperature during a surveillance run

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 five systems/components were adequate to verify system operability and functional capability. The inspectors also witnessed the test or reviewed the test data to verify test results adequately demonstrated restoration of the affected safety functions. Documents reviewed are listed in the Attachment.

- FNP-0-IMP-400.9, Air Operated Valve and Dampers Testing following replacement of cushion regulator for Q2P16FV3009A
- FNP-2-STP-80.1, DG 2B Operability Test following repair of air leak upstream of Q2R43V0595
- FNP-2-SOP-23.0, 2B Component Cooling Water (CCW) heat exchanger service water (SW) discharge flow indicator testing following replacement of instrument root valve Q2P16V268B
- FNP-0-MP-78.0, Governor Speed Setting Adjustment for DG 1-2A, 1B or 2B following repairs to 1-2A EDG governor
- FNP-1-STP-22.7, AFWP Train A Functional Test following repair of the C2 wire for the SGBX relay in the breaker for the 1A MDAFW pump

b. Findings

Introduction: The NRC identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to establish adequate procedures to verify the capability of automatic closure of the Unit 1 and Unit 2 Steam Generator blowdown (SGBD) isolation valves. SGBD isolation valve test procedures did not establish acceptance criteria to verify valve closure in response to an automatic start of the Train A or Train B MDAFW pumps as required by plant design.

Description: The NRC inspectors reviewed licensee CR 2007102460 documenting a broken wire inside the 1A MDAFW pump breaker cubicle. The licensee concluded the broken wire was associated with the SGBD blowdown valve isolation function and did not adversely affect operability of the AFW system. The licensee subsequently repaired the broken wire and returned the AFW system to service 52 hours later. The NRC inspectors questioned the licensee related to accident analysis assumptions for SGBD isolation and the effect on AFW system operability with respect to decay heat removal capability. The licensee generated CR 2007102810 for corporate engineering to evaluate AFW system operability and concluded the 1B MDAFW pump was not affected by the broken wire and remained available to perform the SGBD isolation function.

Farley FSAR Table 3K.3-1 identifies SGBD isolation as required following a high energy line break, and FSAR section 10.4.8.4 states periodic tests of the blowdown isolation valve function will be performed to check operability. The inspectors reviewed the functional system description and the most limiting analysis would require one MDAFW pump to start and deliver 285 gallons per minute (gpm) total flow to all three steam generators. Based on MDAFW pump inservice testing flow rates of approximately 340 gpm and total nominal SGBD flow rates of 90 gpm, the inspectors concluded a failure of the SGBD valves to isolate could reasonably challenge the operability of the AFW system.

The NRC inspectors reviewed completed station test procedures and could not find evidence the design function of automatic closure of the SGBD valves was included as acceptance criteria for the performance of those tests. The inspectors informed the licensee of this issue, who then entered CR 2009107127 in their CAP. Additionally, the inspectors reviewed the licensee's CAP database and discovered CR 2006102769. This

CR documented a Unit 2 condition that required additional testing of the contacts associated with the SGBX relay, which supports automatic start of its associated MDAFW pump and sends an isolation signal to the SGBD blowdown valves. Engineering test procedure ETP-4510 was written to test the SGBX relay in the 2A MDAFW pump automatic start circuitry, including the automatic isolation of the SGBD blowdown valves. The inspectors did not find evidence this procedure was performed, or evidence of the existence of similar procedures to test the 2B, 1B, or 1A MDAFW pumps.

The licensee revised Unit 1 surveillance test procedure, FNP-1-STP-74.0, AMSAC Actuation Test, to include the automatic closure function as required test acceptance criteria on April 29, 2009. The surveillance test was completed satisfactorily a few days later. The licensee produced evidence Unit 1 and Unit 2 surveillance test procedures in use during 2001 documented closure of SGBD isolation valves (although those test procedures did not require valve closure as test acceptance criteria). The inspectors noted during the last automatic reactor trips for each Unit (Unit 1 on November 19, 2008 and Unit 2 on October 3, 2007) that plant computer trend data indicated SGBD flow isolation occurred as a result of MDAFW pump actuation.

Analysis: The failure to establish and implement adequate test procedures including acceptance criteria necessary to verify safety-related equipment is capable of performing its design function is a performance deficiency. This finding is more than minor because the condition could result in the failure to recognize safety function testing acceptance criteria specified by plant design had not been met, and would be a more significant safety concern. The inspectors evaluated the significance of this finding using IMC 0609, Significance Determination Process Attachment 4, Initial Screening and Characterization of Findings Phase 1 worksheets for the mitigating systems cornerstone and determined the finding was of very low safety significance (Green) because it did not result in a loss of safety function of a single train for greater than the TS allowed outage time and was not potentially risk-significant due to external events. This finding is assigned a cross-cutting aspect in the corrective action program component of the PI&R area (P.1(d)) because, although the licensee had identified the need for additional testing of contacts related to SGBD isolation in March 2006, no corrective actions were taken to address this issue in a timely matter, commensurate with safety significance and complexity.

Enforcement: 10 CFR 50, Appendix B, Criterion XI, Test Control, states, in part, that a test program shall be established to assure all testing required to demonstrate components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the NRC determined the licensee failed to implement adequate procedures used to verify the SGBD isolation valves would perform their automatic isolation function. Specifically, station test procedures did not include acceptance criteria to verify the SGBD isolation valves would close in response to an automatic start of the Train A or Train B MDAFW pumps as required by plant design. Because this failure to implement adequate test procedures is of very low safety significance and has been entered into the CAP as CR 2009107127, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000348,364/2009003-01 Inadequate Test Procedures for SG Blowdown Isolation Valves.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

Refueling Activities: The inspectors reviewed the following activities related to the Unit 1 refueling outage (RFO) to verify compliance with 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 TS requirements. 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 Spent Fuel Pool (SFP) Cooling
- Containment heavy load lifts

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the following eight surveillance tests and either observed the test or reviewed test results to verify testing adequately demonstrated equipment operability and met 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 NMP-OS-007, Conduct of Operations and attended selected briefings to determine if procedure requirements were met. Documents reviewed are listed in the Attachment.

Surveillance Tests

- FNP-1-STP-22.30, TDAFW Check Valve Reverse Flow Closure Operability
- FNP-1-STP-38.1, Reactor Trip Manual Handswitch Contact Operability Verification
- FNP-1-STP-256.1E, Reactor Safeguards Response Time Test Instrument Group 5
- FNP-2-STP-45.4, Emergency Core Cooling System (ECCS) Refueling Outage Valves Inservice Test
- FNP-1-STP-16.2, 1B Containment Spray Pump Quarterly Inservice Test
- FNP-1-STP-40.0, Safety Injection With Loss of Off-Site Power Test

In-Service Test (IST)

- FNP-1-STP-11.2, 1B RHR Pump Quarterly Inservice Test

Reactor Coolant System (RCS) Leak Detection

- FNP-1-STP-9.0, RCS Leakage Test

b. Findings

Introduction: A Green self-revealing NCV of TS 5.4.1.a. for failure to implement FNP-1-STP- 33.8, Verification of Steam Dump Arming from Reactor Trip (P-4) Signal and P-4 Contact Verification, was identified when licensee personnel took actions not directed by the procedure. During performance of FNP-1-STP- 33.8, licensee personnel changed the established initial conditions by placing the Solid State Protection System (SSPS) into a different operating mode.

Description: On April 16, the licensee began performance of FNP-1-STP-33.8. The performance of this procedure was scheduled when the SG's were filled; however, problems with the reactor trip breakers (RTBs) delayed performance for several days. Consequently, plant conditions on April 16 were such that the steam generators had been drained below SG LO-LO level reactor trip input to SSPS. One of the initial conditions for this procedure was that no unblocked reactor signals were present. During the pre-job briefing, the licensee recognized that the SG LO-LO level reactor trip was present. After discussions with maintenance and instrumentation and control personnel, it was decided this initial condition was met because both SSPS train A and train B were in the BYPASS mode ("Normal Method") in accordance with FNP-1-IMP-0.7, Disable SSPS Outputs For Modes 5 and 6.

Step 5.3.16 directed that RTB 'A' be closed from the main control board. When the control room operator attempted to close RTB 'A,' the breaker did not close. Licensee personnel at the RTB 'A' cubicle reported that the breaker attempted to close then immediately tripped open. After additional discussions, the licensee determined that having SSPS outputs disabled using the "Normal Method" did not block the SG LO-LO level reactor trip input to SSPS. The licensee decided to change the operating mode of SSPS to inhibit inputs to both SSPS train A and train B ("Alternate Method") in accordance with FNP-1-IMP-0.7. This action was not directed by FNP-1-STP-33.8 and changed the initial conditions previously established for FNP-1-STP-33.8. With the SSPS in the "Alternate Method," RTB 'A' was successfully closed.

At step 5.3.32, the SSPS train ‘A’ logic cabinet output voltage was checked to be less than or equal to 3 VDC. The output voltage was found to be 48.02 VDC. The licensee discussed this condition and determined that the cause may be due to opening RTB ‘A’ locally at step 5.3.28. This step did not specify if RTB ‘A’ was to be opened locally or from the main control board. The licensee decided to take the main control board handswitch for RTB ‘A’ to the TRIP position. Step 5.3.32 was re-performed, but there was no change in the output voltage. At this point, the licensee determined the procedure could not be completed and restored SSPS alignment. A subsequent review by the licensee determined that placing SSPS in the “Alternate Method” allowed RTB ‘A’ to be closed; however; the SSPS logic cabinet output voltage with the RTB open will not be less than 48 VDC.

The inspectors interviewed licensee personnel involved with the performance of the procedure. During the interview, licensee personnel stated that they were informed this procedure needed to be completed due to earlier delays with the refueling outage and schedule constraints were a factor in their decision to change SSPS into a different operating mode.

Analysis: The failure to follow FNP-1-STP- 33.8 is a performance deficiency. This finding is more than minor because it was associated with the Mitigating Systems cornerstone attribute of Configuration Control and adversely affected cornerstone objective in that failure to follow the procedure resulted in changing the initial conditions previously established. This finding was assessed using IMC 0609, Appendix G, Shutdown Operations Significance Determination Process Phase 4 checklist for RCS level >23’OR PWR Shutdown Operation with Time to Boil >2 hours And Inventory in the Pressurizer and determined to be of very low safety significance because the finding did not increase the likelihood of the loss of RCS inventory, degrade the licensee ability to terminate a leak path, and the procedure was successfully preformed prior to entering a mode that required the SSPS to be operable. This finding has a cross-cutting aspect of Work Control in the area of Human Performance (H.2(b)) in that the licensee did not keep personnel apprised that the SG LO-LO level reactor trip signal into SSPS would not be blocked using the “Normal Method” for disabling SSPS.

Enforcement: Technical Specification 5.4.1.a stated, in part, that procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, shall be implemented. RG 1.33, Revision 2, Appendix A, recommended procedures for surveillance testing of the reactor protection system. Contrary to the above, on April 16, 2009, procedure FNP-1-STP- 33.8 was not implemented as written in that licensee personnel placed SSPS in a different operating mode. Placing SSPS in a different operating mode was not specified by the procedure and changed the initial conditions that were previously established for the procedure. Because this violation was determined to be very low safety significance (Green) and was entered into the licensee’s CAP as CR 2009104718, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000348,364/2009003-02 Failure to Follow Procedure.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The NRC evaluated the conduct of routine licensee emergency drills on the following dates to identify any weaknesses and deficiencies in classification, notification and protection action recommendation (PAR) development activities. The NRC observed emergency response operation in the simulated control room to verify event classification and notifications were performed in accordance with FNP-0-EIP-9.0, Emergency Classification and Actions. The NRC used procedure FNP-0-EIP-15.0, Emergency Drills, as the inspection criteria. The NRC also evaluated the licensee critique of the drill to compare any inspector-observed weaknesses with those identified by the licensee in order to verify whether the licensee was properly identifying issues.

- May 13 – SG Tube Leak Coincident with Fuel Cladding Failure and Subsequent Large Break Loss of Coolant Accident

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls. Licensee activities for controlling and monitoring worker access to radiologically significant areas and tasks were evaluated. The inspectors evaluated changes to and adequacy of procedural guidance; directly observed implementation of established administrative and physical radiological controls; appraised radiation worker (radworker) and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection activities; and assessed occupational exposures from radiation and radioactive material.

The inspectors directly observed controls established for workers and HPT staff in airborne radioactivity area, radiation area, high radiation area (HRA), locked-high radiation area (LHRA), and very high radiation area (VHRA) locations. Controls and their implementation for HRA keys and for storage of irradiated materials within Unit 1 (U1) and Unit 2 (U2) spent fuel pool (SFP) areas were reviewed and discussed in detail. The inspectors reviewed and evaluated select U1 Refueling Cycle 22 Outage (1R22) tasks including reactor vessel head disassembly; reactor vessel sump neutron dosimetry change-out, transfer cart/canal inspection/ maintenance; fuel off-load, and fuel inspection activities. As applicable, the inspectors attended pre-job briefings and reviewed radiation work permit (RWP) details to assess communication of radiological control requirements to workers. Occupational workers' adherence to selected RWPs and HPT proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Direct alarming dosimeter (DAD) alarm set points and worker stay times were evaluated against area radiation survey results and actual dose rates encountered and doses received. Worker exposure as measured by DAD, or by licensee evaluations of potential skin doses resulting from discrete radioactive particle (DRP) and dispersed skin contamination events during conduct of 1R22 activities were reviewed and assessed independently. For HRA tasks involving potentially significant dose rate gradients, e.g., insulation removal, transfer canal/ cart maintenance activities, and reactor vessel neutron dosimetry replacement, the inspectors evaluated the potential use of dosimeter multi-badging to monitor worker exposure.

Postings for access to radiologically controlled areas (RCAs) and physical controls for the U1 reactor building containment (RB) and for U1 and U2 reactor auxiliary building (RAB) locations designated as LHRAs and VHRAs were evaluated during facility tours. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys and results for the U1 RB, and for U1 and U2 RAB equipment and work locations, and Low-level radioactive waste and material storage areas. All results were compared to current licensee surveys and assessed against established postings and radiological controls. For the licensee's Independent Spent Fuel Storage Installation (ISFSI), the inspectors reviewed monitoring data and evaluated RWP controls, and postings.

Based on review of selected condition reports, radiation surveys, and discussions with cognizant licensee representatives, the inspectors also evaluated radiological controls for previous U2 Refueling Cycle 19 Outage activities including controls for the U2 cavity drain line, bottom mount inspections, and charging pump maintenance. In addition, the inspectors evaluated controls and licensee response for a February 7, 2009, reactor coolant system filter change-out which resulted in extensive area and personnel contaminations.

Licensee controls for airborne radioactivity areas with the potential for individual worker internal exposures of greater than 30 millirem Committed Effective Dose Equivalent were evaluated. For selected RWPs identifying potential airborne and highly contaminated areas associated with 1R22 activities, e.g., under vessel inspections, valve maintenance, and cavity deconning activities, the inspectors evaluated the implementation and effectiveness of administrative and physical controls including air sampling, alpha-monitoring, barrier integrity, engineering controls, and postings. Licensee identification and assessment of potential occupational radionuclide intakes between January 1, 2008, through April 9, 2009, were reviewed and evaluated.

Radiation protection activities were evaluated against Updated Final Safety Analysis Report (UFSAR) details, and against Technical Specification (TS), and 10 Code of Federal Regulations (CFR) Parts 19 and 20 requirements. Specific assessment criteria included UFSAR Section 12, Radiation Protection; 10 CFR 19.12; 10 CFR 20 Part 20; TS Sections 5.4, Procedures, and 5.7, High Radiation Areas; and approved procedures.

Detailed procedural guidance and records reviewed for this inspection area are listed in Section 2OS1 of the report Attachment.

Problem Identification and Resolution Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Nuclear Management Procedure (NMP)-GM-002, Corrective Action Program, Version (Ver.) 7, and NMP-GM-002-001, Corrective Action Program Instructions, Ver. 8. Licensee Condition Report (CR) documents and audits associated with access controls, personnel monitoring instrumentation, and personnel contamination events were reviewed. Licensee CAP documents reviewed and evaluated in detail during inspection of this program area are identified in Section 2OS1 of the report Attachment.

The inspectors completed 21 of the specified line-item samples detailed in Inspection Procedure (IP) 71121.01. In addition, the inspectors evaluated radiation protection activities detailed in IP 60855 for the licensee's ISFSI facility.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee data for the Performance Indicators (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. 5, was used to verify the basis in reporting for each data element. Documents reviewed are listed in the Attachment.

Cornerstone: Mitigating Systems

- Mitigating System Performance Index, Emergency AC Power
- Mitigating System Performance Index, Heat Removal
- Safety System Functional Failure (SSFF)

The inspectors reviewed samples of raw PI data, LERs, and Monthly Operating Reports for the period covering April 2008 through March 2009. 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 the PI data was appropriately captured for inclusion into the PI report, as well as ensuring the individual PIs were calculated correctly.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Daily CR Reviews

As required by Inspection Procedure 71152, Identification and Resolution of Problems and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the NRC performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing hard copies of CRs, attending daily screening meetings and accessing the licensee's computerized database.

.2 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the two issues listed below for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of CRs and (7) completion of corrective actions in a timely manner.

- Degradation of Auxiliary Building Water-tight Doors
- SW Dye Flow Testing to meet quarterly IST requirements

b. Findings:

Introduction: The inspectors identified a Green, NCV of 10 CFR 50.65(b). Auxiliary Building water-tight doors not included within the scope of the MR monitoring program.

Description: During routine plant inspections, the inspectors noted degraded door seals and worn hinges on water-tight doors located in the Auxiliary Building. The inspectors reviewed the licensee's CR database and MR scoping manual. From this review, the inspectors discovered CRs for the 1B MDAFW and the CCW heat exchanger room water-tight doors (Doors 176 and 168 respectively) which had issues with hinges. Door 176 would not close and Door 168 was difficult to open or close. The inspectors interviewed plant personnel to determine the MR status of these water-tight doors. From these interviews and review of the MR scoping manual, the inspectors determined these doors were not included in the Auxiliary Building function (V15-FO1) of providing a structure to house equipment. Section 3.3 of the licensee's MR scoping manual states, in part, that non-safety related SSCs relied upon to mitigate accidents or transients must be included in the scope of the MR. The inspectors evaluated the accessible doors in each unit to determine the integrity of these watertight doors. Seven doors in Unit 1 and nine doors in Unit 2 were evaluated. In addition to the hinge problems, the inspectors discovered two doors in Unit 1 with degraded seals (excessive hardening of the rubber seal with small chunks missing) and one door on Unit 2 where the seal area had cracks completely across. The inspectors determined the sealing function assumed in the licensee internal flooding analysis for these doors was challenged. The inspectors also concluded the material condition of multiple water-tight doors were such that effective control of performance or condition through the use of appropriate preventative maintenance could not be demonstrated.

Analysis: The inspectors determined the licensee's failure to include the water-tight doors within the scope of their MR program and the subsequent degraded condition of these doors was a performance deficiency. The performance deficiency is more than minor because it adversely affected the equipment performance attribute of the Mitigating System Cornerstone objective to ensure the availability, reliability, and capability of systems responding to initiating events to prevent undesirable consequences (i.e. core damage). The inspectors evaluated the significance of this finding using IMC 0609, Significance Determination Process Attachment 4, Initial Screening of this Characterization of Findings Phase 1 worksheets for the mitigating systems cornerstone and determined the finding was of very low safety significance (Green) because it did involve the degradation of equipment or function specifically designed to mitigate a flooding event (i.e. flooding barriers), but did not result in a loss of safety function. The inspectors did not identify a cross-cutting aspect because the finding is not reflective of current licensee performance.

Enforcement: 10 CFR 50.65(b) which states, in part, that the scope of the monitoring program shall include non-safety related SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function. Contrary to the above, the licensee did not include the water-tight doors for the MDAFW pump rooms, HHSI pump rooms, Containment Spray pump rooms, RHR pump rooms in their MR monitoring program. Failure of these doors could prevent safety-related SSCs from fulfilling their safety-related function. Because this issue is of very low safety significance and is entered into the licensee's CAP as CR2009106669, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000348,364/2009003-03 Failure to Include Water Tight Doors in the Scope of the MR.

c. Observations:

The inspectors noted the licensee performed multiple dye flow tests of Unit 1 Train 'A' SW System during the week of May 10, 2009, and declared the 1A SW Pump inoperable due to excessive flow rates. A similar occurrence was also noted on March 16, 2009. Multiple dye flow tests of Unit 1 Train 'B' SW System were performed over the weekend and during the week resulting in the licensee declaring 1D and 1C SW Pump inoperable. The inspectors interviewed station personnel and reviewed licensee CRs identifying testing inconsistencies. The inspectors reviewed station procedure FNP-0-IMP-212.2, SW Pump Test With Fluorescent Dye Dilution, Version 18.0 for the controls related to testing methodology. The inspectors compared station procedure FNP-0-IMP-212.2 with PTC-18-2002 which describes dye flow testing per ASME code requirements.

The inspectors reviewed the data from the previous two years of quarterly testing and did not discover any anomalies with the test data. All data was within the specified accuracy range required by the ASME code specified in document PTC-18-2002. The inspectors noted FNP-0-IMP-212.2 guidance was more general than guidance in PTC-18-2002, but the licensee's test method provided the required information to adequately determine SW pump degradation. The inspectors noted the licensee was performing multiple tests and not resolving why one set of data was more reflective of pump performance than the other sets taken at the same time prior to returning the affected pump to operable status. The inspectors reviewed CR 2009102798 and its action items including vendor recommendations for improvement of the licensee dye flow testing methodology. These action items properly addressed extent of condition, generic implications, and previous occurrences. The inspectors determined the licensee properly classified and prioritized the resolution of the problem; identified root and contributing causes and planned completion of corrective actions in a timely manner.

.3 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 reviewed repetitive equipment and corrective maintenance issues and also considered the results of daily inspector CAP item screening discussed above. The review also included issues documented outside the normal CAP process, including system health reports, corrective maintenance WOs, component status reports, and MR assessments. The inspectors' review nominally considered the six-month period of January 1 through June 30, 2009, 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 the sample of the issues identified in the licensee's trend report were reviewed for adequacy. Documents reviewed are listed in the Attachment.

b. Assessment and Observations

No findings of significance were identified. The inspectors noticed the continuance of a negative trend in the area of procedure use and adherence. Section 1R22 and Section 4OA7 of this report documents three occurrences of failing to follow procedures. The inspectors also noted Southern Company Fleet Oversight audits document minor issues related to procedural adherence.

.4 Operator Work-Around Annual Review

a. Inspection Scope

The inspectors performed a detailed review of the work-around lists for Unit 1 and 2 in effect on May 18, 2009. 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 FNP-0-ACP-17.0, Work-Around Program.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours. These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Findings

No findings of significance were identified.

.2 NRC Temporary Instruction (TI) 2515/172, RCS Dissimilar Metal Butt Welds (DMBWs)

a. Inspection Scope

From April 13 – 17, 2009, the inspector reviewed the licensee's activities related to the inspection and mitigation of DMBWs in the RCS to ensure the licensee activities were consistent with the industry requirements established in the Materials Reliability Program (MRP) document MRP-139, Primary System Piping Butt Weld Inspection and Evaluation Guidelines, July 2005.

TI 2515/172 was performed in 2008 as documented in Inspection Report 2008005. During that time, a complete program review (per TI 2515/172 paragraph 03.05) was performed.

b. Findings and Observations

No findings of significance were identified.

MRP-139 Baseline Inspections

1) Have the baseline inspections been performed or are they scheduled to be performed in accordance with MRP-139 guidance?

Yes. The licensee has performed all required baseline inspections at the time of this review.

No follow-on exams occurred since the baseline inspections had been performed, and based on the categorization of the welds in the program, no follow-on exams were required to have been completed at the time of the inspection.

Therefore, the licensee has met the MRP-139 deadlines for baseline examinations of all welds scoped into the MRP-139 program.

2) Is the licensee planning to take any deviations from MRP-139 requirements?

No, the licensee has not submitted any requests for deviation from MRP-139 requirements.

Volumetric Examinations

Sample not available.

Weld Overlays

Sample not available.

Mechanical Stress Improvement (Not Applicable)

Sample not available.

In-service Inspection Program

This reporting requirement was addressed previously in inspection report 2008005; no new information was noted during this inspection.

4OA6 Meetings, Including Exit

On July 09, 2009, the NRC presented the inspection results to members of your staff who acknowledged the findings. The NRC confirmed proprietary information was not provided or examined during the inspection. An exit meeting for the ISI portion was conducted on April 17, 2009 with licensee management.

On April 10, 2009, the NRC presented the results of the onsite radiation protection inspection with Ms. C. Collins, Plant Manager and other responsible staff. During a June 23, 2009, teleconference with Mr. C. Peters, the inspectors discussed the elevated dose rate areas found during the 1R22 bottom nozzle inspections and the adequacy of licensee radiation protection practices and established controls. Based on review of applicable fuel handling activities and radiation protection procedure implementation, resultant survey data and the licensee's apparent cause determination, the inspectors noted that no radiation control performance deficiencies were identified with the bottom nozzle inspection and no findings of significance were identified.

4OA7 Licensee Identified Violations

The following violations of very low safety significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI.A.1 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 5.4.1, Procedures, require in part that written procedures be implemented for those activities recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A. RG 1.33, Appendix A, Section 3, Procedures for Startup, Operation, and Shutdown of Safety-Related PWR Systems, recommends procedures for maintaining containment integrity. Procedure FNP-1-STP-18.4, Containment Mid-Loop and/or Refueling Integrity Verification and Containment Closure, requires the penetration data sheet be updated when the status of an isolated penetration is changed and the change should be entered in the containment integrity tracking index sheet. Contrary to the above, the containment penetration data sheets for penetrations 4, 5, and 6 were not updated and the change was not entered in the containment index tracking sheet between April 4 - April 10, 2009. The AFW to 1A, 1B, and 1C Steam Generator Stop Valves were required to be closed for containment integrity prior to the movement of irradiated fuel, but were opened to support SG filling. Consequently, on April 11, 2009, during movement of irradiated fuel, the licensee discovered these valves were not in the required closed position for containment integrity. The licensee immediately halted fuel movement and closed these valves prior to resuming fuel movement. The licensee entered this condition into their CAP as CR 2009104301. This violation was evaluated using phase I screening worksheets and was also discussed with both a regional and a headquarters SRA and determined to be of very low safety significance (GREEN) because the associated penetrations were isolated by other closed manual valves in the AFW discharge flow path and the AFW discharge piping remained water filled.

- TS 5.4.1, Procedures, require in part that written procedures be implemented for those activities recommended in RG 1.33, Revision 2, Appendix A. RG 1.33, Appendix A, Section 3, Procedures for Startup, Operation, and Shutdown of Safety-Related PWR Systems, recommends procedures for the CCW and SW systems. Station procedure FNP-2-SOP-23.0, CCW System, was changed on June 8, 2009 to include a section for swapping the on-service CCW heat exchanges with both SW flow control valves on their jacking devices and the valves fully open. This section requires that the on-coming flow control valve (FCV-3009C) be jacked fully open. Contrary to the above, the licensee placed the 2C CCW heat exchanger in-service on June 9 with FCV-3009C jacking device not engaged. Approximately 1.5 hours later, the control room staff noted SW flow through the 2C CCW heat exchanger had decreased to 3300 gpm which is below the required flow for accident conditions. The licensee discovered the undesired condition and restored proper flow. The inoperable condition of CCW existed for 6 hours 39 minutes. The licensee entered this condition into their CAP as CR 2009107473. This finding was assessed using Inspection Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet for mitigating systems cornerstone and determined to be of very low safety significance (GREEN) because the finding does not represent an actual loss of a safety function or loss of one or more trains for more than the allowed TS outage time.

- 10 CFR 50, Appendix B, Criterion III, Design Control, requires in part that measures shall be established to assure the design basis for those SSCs to which this appendix applies are correctly translated into procedures and instructions. Contrary to this on September 24, 2008, the licensee determined that surveillance test procedure FNP-0-STP-123.0, Control Room Emergency Ventilation Performance Test, contained an incorrect formula constant for the cross sectional area of the control room ventilation duct used to establish the required flow conditions for performance of the test. Specifically, the test procedure utilized a constant for the cross sectional duct area that was 0.42 square feet when actual cross sectional duct area was 0.39 square feet. This error resulted in the performance of a surveillance test documented in work order W00703732 where actual system flow rate was lower than that required for performance of the test. The licensee determined surveillance test procedures FNP-0-STP-123.3, CREFS Pressurization Unit Heater Performance Test and FNP-0-STP-916.0, CREFS Pressurization Unit Heater Operability Test, also referenced the incorrect duct cross sectional area and were similarly affected. This was identified in the licensee's CAP as CR 2008109679. The licensee was able to demonstrate through subsequent analysis for those tests performed at lower than recorded system flow rates, sufficient margin remained to support system operability. This finding was assessed using Inspection Manual Chapter 0609 "Significance Determination Process" Phase 1 screening worksheet for mitigating systems cornerstone and determined to be of very low safety significance (GREEN) because it did not result in an actual loss of safety function of a single train for greater than the TS allowed outage time and was not potentially risk-significant due to external events.

- 10 CFR 50 Appendix B, Criterion V, Instructions Procedures, And Drawings, requires in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstance. Contrary to the above on April 22, 2009, the licensee mistakenly stopped all RHR Pumps using procedure FNP-1-STP-40.0, Safety Injection with Loss of Off-site Power Test. Specifically, step 5.27.3 did not provide adequate guidance to ensure at least one RHR pump remained in service with fuel in the reactor vessel. The licensee entered the issue into the CAP as CR 2009105141. This finding was assessed using Inspection Manual Chapter 0609 Significance Determination Process, Phase 1 screening worksheet for mitigating systems cornerstone and determined to be of very low safety significance (GREEN) because the interruption of forced cooling was permitted by TS 3.9.4 and forced cooling was restored within the allowed time frame.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
J. Agold, Southern Nuclear Corporate ISI Program Manager
M. Caldwell, CCW System Engineer
C. Collins, Plant Manager
J. Cox, Southern Nuclear Corporate SW HX Program Manager
M. Dove, Southern Nuclear Corporate Alloy 600 Program Manager
M. Goocher, SW System Engineer
B. Grinder, Engineering Support Manager
P. Hayes, Engineering Director
L. Hogg, Security Manager
J. Horn, Site Support Manager
J. Jerkins, Performance Improvement Senior Engineer
J.R. Johnson, Site Vice President
M. Johnston, ISI Coordinator
T. Livingston, Chemistry Manager
G. Lofthus, Southern Nuclear Corporate Level III
H. Mahan, Licensing Engineer
R. Martin, Technical Services Manager
L. McKay, BACCP Owner
B.D. McKinney, Licensing Supervisor
C. Medlock, Site Design Manager
B.L. Moore, Site Support Manager
K. Moore, Equipment Reliability Supervisor
D. Morrow, Engineering Support Program Supervisor
R. Mullins, Southern Nuclear Corporate Inspection Supervisor
W. Oldfield, Fleet Oversight Supervisor
C. Peters, HP Manager
R. Retherford, Engineering Support (Acting Supervisor)
G. Terry, Southern Nuclear Corporate HX/Cooler Eddy Current Testing Program Manager

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

LIST OF REPORT ITEMS

Opened
None

Opened and Closed

05000348,364/2009003-01	NCV	Inadequate Test Procedures for SG Blowdown Isolation Valves (Section 1R19)
05000348,364/2009003-02	NCV	Inadequate Test Procedures for Verification of Steam Dump Arming From Reactor Trip (P-4) Signal and P-4 Contact Verification (Section 1R22)

05000348,364/2009003-03	NCV	Failure to Include Water Tight Doors in the Scope of the Maintenance Rule (Section 4OA2)

<u>Closed</u>
None

<u>Discussed</u>
None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures:
FNP-0-ACP-4.0, Switchyard Control, Version 9.0
FNP-0-AOP-21.0, Severe Weather, Version 25.0
FNP-1-AOP-5.2, Degraded Grid, Version 12.0
FNP-1-UOP-3.1, Power Operation, Version 100.0
FNP-2- AOP-5.2, Degraded Grid, Version 12.0
FNP-2-UOP-3.1, Power Operations, Version 85.0

Section 1R04: Equipment Alignment

Documents:
TS 3.4.8, RCS Loops – MODE 5, Loops Not Filled
D-175038, P&ID – Safety Injection System, Sheet 2, Version 21.0
A-181002, Functional System Description Residual Heat Removal/Low Head Safety Injection, Version 37.0

Procedures:
FNP-1-AOP-12.0, Residual Heat Removal System Malfunction, Version 22.0
FNP-1-SOP-7.0, Residual Heat Removal System, Version 84.0

Section 1R05: Fire Protection

Plant Drawings:
A-508650, Sheet 12, Version 2.0
A-508650, Sheet 14, Version 2.0
A-508650, Sheet 27, Version 1.0
A-508650, Sheet 33, Version 1.0
A-508650, Sheet 46, Version 3.0
A-508650, Sheet 47, Version 3.0
A-508650, Sheet 48, Version 1.0
A-508650, Sheet 49, Version 1.0

Section 1R08 & 4OA5: Inservice Inspection (ISI) Activities

Calculations
FNP-0-ETP-4496, Boric Acid Leak Evaluation for CR 2009103927, dated 04/10/2009
FNP-0-ETP-4496, Boric Acid Leak Evaluation for CR 2009103966, dated 04/08/2009
FNP-0-ETP-4496, Boric Acid Leak Evaluation for CR 2009103971, dated 04/08/2009
Westinghouse Loose Parts Evaluation for Farley Steam Generator #1, dated 04/16/2009
IER 1R21 002, NDE Indication Evaluation Report for ALA1-1100-7,-18,-19,-20,-21,-22, dated 10/17/07

Corrective Action Documents
CR 2009103966, Dried Boron on Pressurizer, dated 04/07/2009
CR 2009104690, Boric Acid Leak/Spill from CETNA at Penetration 40, dated 04/15/2009
*CR 2009104754, Effect of Cleaning Boric Acid from Reactor Pressure Vessel Head, dated 04/16/2009
*CR 2009104786, Compartmentalized Communications Issues, dated 04/16/2009

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Procedures
FNP-0-ETP-4496, Corrosion Assessment, Version 2.0
FNP-0-M-101, Boric Acid Corrosion Control Program, Version 13
FNP-2-STP-34.0, Containment Inspection (General), Version 22.0
NMP-ES-019, Boric Acid Corrosion Control Program, Version 5.0
NMP-ES-019-001, Boric Acid Corrosion Control Program Implementation, Version 3.0
NMP-ES-024-202, Visual Examination (VT-2), Version 2.0
NMP-ES-024-203, Visual Examination (VT-3), Version 2.0
NMP-ES-024-208, Visual Examination of Reactor Vessel Head penetrations and Base material (Remote and Direct), Version 2.0
NMP-ES-024-301, Liquid Penetrant Examination Color Contrast and Fluorescent, Version 5.0
NMP-ES-024-501, PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds (Appendix VIII), Version 2.0

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F1 ALA1-4101-8, Component Summary for UT of ALA1-4101-8 Pipe-to-Valve, dated 4/9/09
F1 ALA1-4101-RHR-R99 (W2), Component Summary for PT of ALA1-4101-RHR-R99 (W2), dated 4-16-09
F1 ALA1-1300-BMV, Component Summary for Bare Metal Visual Exam of RPV Closure Head Outside Surface, dated 4-21-09
Magnaflux Form F-1569, Certification for PT Spotcheck Developer, Batch 05E12K, Rev. 5/06
Magnaflux Form F-1579E, Certification for PT Spotcheck Cleaner/Remover, Batch 07D01K, Rev. 12/06
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Nutley Equipment Repair Certification of Calibration for Light Intensity Meter, dated January 19, 2009
PDI Certification for Ultrasonic Testing (DiValerio, Grell, VanRuler, Washburn), various dates
PQR B05, Procedure Qualification Record, dated 10-5-76 and 11-8-82
PQR B06, Procedure Qualification Record, dated 10-6-76 and 11-8-82
PQR B077, Procedure Qualification Record, dated 03/25/87 and 4/27/87
PTC Metrology Certification No. 10432, Thermometer Calibration Certification, dated 01/06/09
VT/PT Certification & Eye Exams for NDE Personnel (Coburn, Huls, Lofthus), various dates
WPS 8.20N, Weld Procedure Specification, Rev. 6
WPS 8.22N, Weld Procedure Specification, Rev. 5
WO# 1090687701, Sock-o-Let Repair for SW Supply to 1C Containment Cooler Drain, Rev. 1
WO# 1082199502, Installation of Vent Valves in Response to GL 2008-01, Rev. 0
WO# 1082199503, Installation of Vent Valves in Response to GL 2008-01, Rev. 0

Section 1R11: Licensed Operator Requalification

Condition Reports:
2009103023, 2009103029

Documents:
OPS-5600A, Licensed Operator Continuing Training Simulator Exercise Guide, Scenario 09-S602, dated 4/30/2009

Procedures:
FNP-1-AOP-8.0, Partial Loss of Condenser Vacuum, Version 20.0
FNP-1-AOP-3.0, Turbine Trip Below P-9 Setpoint, Version 16.0
FNP-1-AOP-17, Rapid Load Reduction, Version 19.0
FNP-1-EEP-0, Reactor Trip or Safety Injection, Revision 37
FNP-0-EIP-9.2, Emergency Classification, Version 7.0

Section 1R12: Maintenance Rule Effectiveness

Condition Reports:
2005100545, 2005105688, 2005105774, 2005111338, 2005111529, 2005111797, 2005112395, 2005112444, 2006102769, 2006104453, 2006104655, 2007103267, 2007109394, 2007109554, 2007109885, 2007109892, 2007109938, 2007111426, 2008111182, 2008111986, 2008112298, 2008112299, 2008112300, 2008112433, 20081113395, 2009103892, 2009100875, 2009106295, 2009106304, 2009107473, 2009107507

Documents:
Action Item: 2009200926

Procedures:
FNP-0-GMP-27.0, Disassembly and Reassembly of Safety Related and Non-Safety Related Valves, Version 24.0
FNP-0-GMP-27-5, Valve Packing Replacement Safety Related And Non-Safety Related Valves, Version 23.0

Work Orders:
1090199201, 2071070801, 2082092201, 208218601, 2082247301, M200590901, M400123901, M00289701, S400318601

Section 1R15: Operability Evaluations

Condition Reports:
2009103797, 2009103799, 2009105601, 2009106789, 2009107292

Documents:
DOEJ-FX-2008103938-M001, Evaluation of CVCS Pump Casing Leak Impact on Offsite and Control Room Doses
DOEJ-SM-2070129101-003, Unit 2 Low-Head Safety Injection Containment Sump Suction Isolation Valve Encapsulation Configuration

RER 2070129101, Q2E11MOV8811A Encapsulation
TS 3.7.4, Atmospheric Relief Valves (ARVs)

Procedures:
FNP-0-SOP-0.0, General instructions to Operations Personnel, Version 124.0
FNP-2-SOP-62.0, Emergency Air Systems, Version 22.0
FNP-2-SOP-17.0, Main and Reheat Steam, Version 49.1

Section 1R19: Post Maintenance Testing

Condition Reports:
2008107536, 2008112346, 2008113828, 2009103473, 2009106676, 2009104954, 2009105048, 2007102810, 2007102460, 2006102769

Action Items:
2006201275, 2006201531

Documents:
D-200212, P&ID – Air Start System for Diesel Generator 2B, Version 23.0
D-205003, P&ID – Service Water System, Version 36.0
D-207186, Elementary Diagram – Auxiliary Feedwater Pump 4160V No. 2A, Version 15.0
D-207394, Elementary Diagram – Multiplying Relay Cabinets Train “A”, Revision 21
D-207385, Elementary Diagram – Solenoid Valves, Version 8.0
A-181010, Functional System Description – Auxiliary Feedwater (AFW) System, Revision 9.0
TS 3.7.5, Auxiliary Feedwater (AFW) System

Procedures:
FNP-0-IMP-400.9, Air Operated Valve and Dampers Testing, Version 14.0
FNP-0-MP-78.0, Governor Speed Setting Adjustment for Diesel Generator 1-2A. 1B or 2B, Version 7.0
FNP-2-STP-80.1, Diesel Generator 2B Operability Test, Version 43.0
FNP-2-SOP-23.0, Component Cooling Water System, Version 78.0
FNP-1-STP-22.7, Auxiliary Feedwater Pump Train A Functional Test, Version 28.0
FNP-2-ETP-4510, Auxiliary Feedwater Pump 2A Auto Start Test, Version 1.0
FNP-1-STP-74.0, AMSAC Actuation Test, Version 23.0

Work Orders:
2082215501, 2082477801, 2090998601, S081526901, 1070670301, 1060372801, 2053128701, 2060951501, 2060927001

Section 1R20: Outage Activities

Procedures:
FNP-0-UOP-4.0, General Outage Operations Guidance, Version 33.0
FNP-1-UOP-4.1, Controlling Procedure for Refueling, Version 49.0
FNP-0-AP-94, Outage Nuclear Safety, Version 8.0
FNP-1-UOP-4.3, Mid-Loop Operations, Version 27.0
FNP-0-ACP-47.3, Outage Preparation, Version 13.0

FNP-1-STP-35.0, Reactor Coolant System Pressure and Temperature/Pressurizer Temperature Limits Verification, Version 19.0
FNP-1-UOP-2.1, Shutdown of Unit from Minimum Load to Hot Standby, Version 63.0
FNP-1-UOP-2.2, Shutdown of Unit from Hot Standby to Cold Shutdown, Version 85.0
FNP-1-SOP-1.6, Draining the Reactor Coolant System, Version 51.0
FNP-1-SOP-1.3, Reactor Coolant System Filling and Venting-Vacuum Method, Version 59.0
FNP-1-STP-18.4, Containment Mid-Loop and/or Refueling Integrity Verification and Containment Closure, Version 33.0
FNP-1-IMP-201.45, Refueling Reactor Coolant System Level Loop Calibration Q1B21FT0416, Version 12.0
FNP-1-STP-35.1, Unit Startup Technical Specification Verification, Version 41.0
FNP-0-ETP-3643, Verification of Rod Control System Operability, Version 44.0
FNP-1-STP-101, Zero Power Reactor Physics Testing, Version 13.0
FNP-1-STP-29.6, Calculation of Estimated Critical Condition, Version 8.0
FNP-1-MP-11.4, Reactor Polar Crane - Operating and Safe Load Path Instructions, Version 24.0
NMP-MA-007-009, SNC Rigging and Lifting Program Plant Farley Specifics, Version 2.0
FNP-1-MP-1.0, Maintenance Refueling Procedure, Version 48.0

Section 1R22: Surveillance Testing

Condition Reports:
2008111801, 2009105222, 2009103627, 2009104440, 2009104718, 2009104759, 2009105014, 2009105516, 2009105672, 2009105712, 2009105831, 2009106066, 2009105141

Documents:
Analysis and Measurement Services (AMS) Corporation Rod Movement Results for Farley Unit 2
D207614, Elementary Diagram 575V Motor Operated Valve MOV8803B-AB, Revision 1
GOTHIC Shutdown Analysis to Support Farley Units 1 and 2, dated January 2007
Control room Operating Logs, dated 4/22/2009
Pre-Job Brief for FNP-1-STP-40.0, Safety Injection with Loss of Off-Site Power Test

Procedures:
FNP-1-STP-9.0, RCS Leakage Test, Version 47.0, test performed 5/5/2009
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FNP-1-STP-11.2, 1B RHR Pump Quarterly Inservice Test, Version 51.0
FNP-1-STP-16.2, 1B Containment Spray Pump Quarterly Inservice Test, Version 49.0
FNP-1-STP-256.1E, Reactor Safeguards Response Time Test – Instrument Group 5, Version 21.0
FNP-2-STP-45.4, ECCS Refueling Outage Valves Inservice Test, Version 19.0
FNP-1-STP-40.0, Safety Injection with Loss of Off-Site Power Test, Version 54.1
NMP-OS-007-001, Conduct of Operations Standards and Expectations, Version 4.0
FNP-1-ARP-1.3, Main Control Board Annunciator Panel C, Version 27.0
MP-AD-006, Infrequently Performed Tests of Evolutions, Version 5.0

Work Orders:
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Section 2OS1: Access Controls to Radiologically Significant Areas

Procedures, Manuals, and Guidance Documents

Radiation Work Permit (RWP) 09-0101, Administration: Activities in RCA by ADM, SEC, IT, FO, Safety and Health, NRC, TRN, & Work Control personnel, Revision (Rev.) 0
RWP 09-1461, MAINTENANCE: All work associated with disassembly & reassembly of the Rx Head in support of the Unit 1 Refueling Cycle 22 (1R22) Outage. (Includes work on Sand Box Covers, NI Covers, Stud Hole Plugs, Dirt Barrier, and Cavity Seal Ring), Revision (Rev.) 0 & Rev 2
RWP 09-1467, MAINTENANCE: Cleanout and maintenance in the Rx cavity transfer canal in support of the 1 R22 Outage, Rev. 1
RWP 09-1481, MAINTENANCE: All activities inside regeneration heat exchanger fence & Rx cavity drain valve fence in support of 1 R22 Outage, Caution this RWP is for LHRAs, Rev. 0
RWP 09-1482, MAINTENANCE: Insulation activities in support of 1 R22 Outage, (Does not include insulation activities associated with Rx Head, Rx Vessel Nozzles, Pressurizer, ISI Activities) Rev. 0
RWP 09-1504, OPERATIONS: Refueling activities in support of the 1 R22 Outage, Rev. 0 and Rev. 2
RWP 09-1782, ENGINEERING: Inspection and walkdowns of Rx vessel nozzles in support of the 1R22 outage. CAUTION: This RWP is for locked high radiation areas, Rev. 1
Farley Nuclear Plant (FNP)-0-Radiation Control and Protection Procedure (RCP) 13.1, Use of the HIS-20 RWP Section, Version (Ver.) 19.0
FNP-0-RCP-0.2, Unit 1 Reactor (Rx) vessel maintenance sump entry, Ver. 4
FNP-0-RCP-4, Refueling survey
FNP-0-RCP-29.1, Guidelines for personnel decon and response to personnel contamination events, Ver. 10.0
FNP-0-Fuel Handling Procedure (FHP)-7.0, Limitations and precautions for handling fuel assemblies

Records and Data Reviewed

CY 2009, January 1, 2009, through April 7, 2009, Personnel Contamination Event/ Personnel Contamination Report Data
HP Form 602, Farley Nuclear Plant Health Physics Turnover Logs, 04/06-10/09
Radiological Survey Number (#)50344, U1 Reactor Cavity (1CB155), 04/06/09
Radiological Survey #50362, U1 C RCP Cavity (1CB129), 04/06/09
Radiological Survey Number (#)48680, Filter change-out survey, 02/07/09
Radiological Survey # 48672, #48676, #48677, #48671, U1 Reactor coolant system filter room (1AB139303), 02/07/09
Radiological Survey # 48671, #48678, #48679, #48737, #48783, #48799, #48830, U1 A/B 139' el (1AB139), 2/11/09, 02/07/09, 02/07/09, 02/07/09, 02/09/09, 02/11/09, 02/12/09
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Radiological Survey Number (#)45068 and #46373; U-1 A/B 100 foot elevation , 10/08/08 & 11/04/08
Radiological Survey # 32551, #32911, and 45429, U2 Cavity Drain line (Inside Missile Barrier)

(2 CB 105'), 05/08/07, 05/17/07, and 10/20/08
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Radiological Information Survey # 48977 PCE/PCR, 02/18/09
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Radiological Information Survey # 49140 PCE/PCR, 02/24/09
Area Perimeter Monitoring TLD Data, 1st Half CY 2008 and 2nd Half 2008

Corrective Action Program (CAP) Documents

Farley (F) Fleet Oversight Assessment (FOA) Report – Health Physics (HP)-2008-1, Change Management Process in Health Physics Department
F-FOA Report –Radiation Protection (RP) 2009-1, Radiation Work Practices,
Condition Report (CR) 2009104382, Investigation to determine cause of worker dose rate alarms
in the U1 SFP room during bottom nozzle inspection
CR 2009104343, Contract employee received dose rate alarm of 232 millirem per hour in the U1
SFP room
CR 2009104073, Improper RWP used for pressurizer manway support activities
CR 2009104027, Issuance of TLD prior to completion of NRC Form 4
CR 2009103894, Locked high radiation area controls for installation of sandbox covers in the Rx cavity
CR 2009103612, Personal dosimetry lost both DAD and TLD
CR 2009103832, Radiation protection controls not in accordance with RWP 09-1782
CR 2009103560, 1R22 OLL" Lessons learned for refueling outages, develop re-lamping to support initial containment surveys
CR 2009101465, Control of high radiation level filters
CR 2009101360, Clean area posted as contaminated
CR 2009101351, Contamination of workers and plant flooring
CR 2008112261, Ingestion of radioactive material
CR 2008111308, Possible dilution of air sample data
CR 2008111148, Pipe-fitter received dose rate alarm
CR 2008110914, Worker DAD alarm during bottom mount inspection in vessel maintenance sump
CR 2008111029, Worker received dose rate alarm while disposing of old in-core detector cables in U2 drumming room shield
CR 2008110915, Use of wrong RWP for performing bottom mount inspections
CR 2008110529, Unexpected high level of airborne activity in the 2 'B' charging pump room
CR 2008110379, Hot Spot in Piping Penetration Room, 100 foot (') elevation potentially causing HRA conditions
CR 2008112835, Elevated dose rates on U2 105 ' elevation at cavity drain line inside bio-shield

Section 4OA1: Performance Indicator Verification 71151

Condition Reports:
2008104235, 2008105884, 2008105920, 2008106308, 2008106505, 2008107725, 2008108616, 2008109193, 2008113825, 2009100525, 2008104627, 2008105323, 2008106177, 2008106794, 2008109778, 2008111197, 2008112298, 2008112300, 2008112659, 2008112672, 2008112675,

2009103314, 2008106306, 2008107003, 2008107223, 2008107290, 2008107297, 2008107527, 2008107536, 2008108745, 2008110101, 2008112053, 2008112268, 2009100533, 2009100948

Documents:
Unit 1, selected control room logs from April 1, 2008 through March 31, 2009
Unit 2, selected control room logs from April 1, 2008 through March 31, 2009
Farley Key Performance Indicators Report, dated March 2009
Unit 1, MSPI Heat Removal System Derivation Reports for Unavailability and Unreliability, period ending March 2009
Unit 1, MSPI Emergency AC Power System Derivation Reports for Unavailability and Unreliability, period ending March 2009
Unit 2, MSPI Heat Removal System Derivation Reports for Unavailability and Unreliability, period ending March 2009
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Procedures:
FNP-0-AP-54, Preparation and Reporting of NRC Performance Indicator Data and NRC Operating Data, Version 12.0

Section 4OA2: Identification and Resolution of Problems 71152
Action Item: 2004204509, 2004205528

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Documents:
ASME PTC 18-2002, Hydraulic Turbines and Pump-Turbines Performance Test Codes
CleanAir Engineering letter dated June 4, 2009 Re: Recommendations for Determination of Service Water Flow by Dye Dilution
Farley Document: Service Water Pump Dye Testing Focused Self-Assessment
Ontario Power Generation Document: Flow Measurement Using the Dye Dilution Technique
REA 93-0358, Service Water Pump Flow vs. Temperature Study
Southern Company Service letter dated August 25, 1994 Service Water Pump Flow Measurement Phase III – Instrument Accuracy File: ENG 15 93-0358

Procedures:
FNP-0-IMP-212.2, “Support Procedure for Service Water Pump Test with Fluorescent Dye Dilution,” Version 0
FNP-0-IMP-212.2, “Support Procedure for Service Water Pump Test with Fluorescent Dye Dilution,” Version 1
FNP-0-IMP-212.2, “Support Procedure for Service Water Pump Test with Fluorescent Dye Dilution,” Version 18
FNP-0-IMP-212.2, “Support Procedure for Service Water Pump Test with Fluorescent Dye Dilution,” Version 19
FNP-0-M-87, “Maintenance Rule Scoping Manual,” Version 20.0
FNP-1-STP-24.2, “1C, 1D and 1E Service Water Pump Quarterly Inservice Test,” Version 63.0

FNP-1-STP-24.13, “1C, 1D and 1E Service Water Pump Biennial Comprehensive Pump Test,” Version 21.0

Work Orders: 1070188901, 1070457001, 1070670701, 1070935501, 1071353901, 1071855701, 1081983201, 1090406101