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

January 31, 2006

Southern Nuclear Operating Company, Inc.
ATTN: Mr. L. M. Stinson
Vice President - Farley Project
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000348/2005005, 05000364/2005005, AND 072000042/2005002

Dear Mr. Stinson:

On December 31, 2005, 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 January 5, 2006, with Mr. Randy Johnson 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, two self-revealing findings of very low safety significance were identified both of which were determined to be violations of NRC requirements. Because these violations are of very low safety significance and were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States 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/

Malcolm T. Widmann, Chief
Reactor Projects Branch 2
Division of Reactor Projects

Docket Nos. 50-348, 50-364, and 72-42
License Nos. NPF-2 and NPF-8

Enclosure: Inspection Report 05000348/2005005,
05000364/2005005, and 072000042/2005002
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-348, 50-364, 72-42
License Nos.:	NPF-2, NPF-8
Report Nos.:	05000348/2005005, 05000364/2005005, and 072000042/2005002
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Joseph M. Farley Nuclear Plant
Location:	7388 N. State Highway 95 Columbia, AL 36319
Dates:	October 1- December 31, 2005
Inspectors:	C. Patterson, Senior (Sr.) Resident Inspector J. Baptist, Resident Inspector C. Rapp, Sr. Project Engineer (Sections 1R06 and 1R07) K. Van Doorn, Senior Reactor Inspector (Sections 1R08 and 4OA5.3) L. Lake, Reactor Inspector (Sections 1R08 and 4OA5.3) S. Rose, Sr. Operations Engineer (Section 1R11) M. Maymí, Reactor Inspector (Section 1R12) G. Kuzo, Sr. Health Physicist (Sections 2OS1, 4OA5.1, and 4OA5.2) A. Nielsen, Health Physicist (Sections 4OA1, 4OA5.1, and 4OA5.2) A. Vargas-Mendez, Reactor Inspector (Section 4OA5.1)
Accompanying Personnel:	B. Crowley, Consultant
Approved by:	Malcolm T. Widmann, Chief Reactor Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000348/2005-005, 05000364/2005-005, 072000042/200-5002; 10/01/2005-12/31/2005; Joseph M. Farley Nuclear Plant, Units 1 & 2, Maintenance Risk Assessments and Emergent Work Control, Refueling and Other Outage Activities.

The report covered a three-month period of inspection by resident inspectors, a project engineer, reactor inspectors, an operations engineer, and health physicists. Two Green non-cited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or 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 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 3, dated July, 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- A Green, self-revealing non-cited violation (NCV) of 10 CFR 50.65(a)(4) was identified when the licensee failed to perform a required risk assessment following alignment of the 1-2K and 1-2L 600V load centers to their emergency power (Unit 1) supplies. This resulted in the risk being elevated from a Yellow to an Orange status without senior management concurrence and no additional compensatory actions.

  This finding is more than minor because it impacted the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the cornerstone objective in that the licensee failed to perform a risk assessment following a change in actual plant configuration and did not establish compensatory measures consistent with the elevated outage risk. Based on an Incremental Core Damage Probability Deficit (ICDPD) of less than 1E-6, this finding is of very low safety significance (Green). This finding involved the cross cutting aspect of Human Performance in the area of Personnel for failure to follow procedure FNP-0-ACP-52.3. (Section 1R13)

- A Green, self-revealing NCV of Technical Specification (TS) 5.4.1.a was identified for inadequate procedural controls to maintain reactor coolant system (RCS) water level instruments operable. The lack of adequate procedural controls resulted in one of the required water level instruments being isolated prior to lowering RCS water level.

  This finding is greater than minor because it affected the procedure quality attribute and the Mitigating Systems Cornerstone objective, in that, procedural controls for operability of RCS level instrument were not appropriate to the circumstances. This finding was determined to be of very low safety significance because actual water level was at the level planned for the drain down. This finding involved the cross cutting aspect of Human Performance in the area of Organization in that procedural controls were not adequate. (Section 1R20)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near rated thermal power (RTP).

Unit 2 operated at or near RTP until September 11 when a power coastdown began. The unit was shut down October 15 for a refueling outage and achieved RTP again on December 9.

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

1R01 Adverse Weather Protection

a. Inspection Scope

Seasonal Readiness Review. The inspectors evaluated implementation of the licensee's Cold Weather Contingency procedure FNP-0-SOP-0.12 and conditions for entry into the procedure. The inspectors inspected protective coverings on the Main Steam valve rooms grating, on circulating water piping, and heat tracing lines on the condensate storage tanks, reactor makeup water storage tanks, and refueling water storage tanks (RWSTs) to verify these protections for cold weather conditions were functional.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walk-downs. The inspectors performed partial walk-downs of the following three systems to verify they were properly aligned when redundant systems or trains were out of service. 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.

- 2A Residual Heat Removal (RHR) pump while the 2B RHR was out of service.
- 1B Emergency Diesel Generator (EDG) during FNP-2-STP-80.14, 'A' Train Loss of Off-Site Power Test.
- 2B Boric Acid Tank (BAT) and supply system while 2A BAT was out of service for bladder replacement.

Complete Walk-down. The inspectors conducted a complete walk-down of the accessible portions of the Unit 2 High Head Safety Injection (HHSI) system. The inspectors used licensee procedures FNP-1-SOP-8.0, High Head Safety Injection System; FNP-1-SOP-8.1A HHSI System Check Lists; and drawings D-205038 and D-205039, to verify adequate system alignment of on-service equipment. The inspectors also interviewed personnel and reviewed control room logs, Maintenance Rule (MR) monthly reports, condition reports (CRs), quarterly system health reports, outstanding work orders, and industry operating experience to verify that alignment and equipment discrepancies were being identified and appropriately resolved. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Fire Area Tours. The inspectors conducted a walk-down of the nine fire areas listed below to verify the licensee's control of transient combustibles, the operational readiness of the fire suppression system, and the material condition and status of fire dampers, doors, and barriers. The requirements were described in 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.

- Units 1 and 2 EDG Building, Switchgear Room Train A, Zone 56A
- Units 1 and 2 EDG Building, Fire Protection Clothing Locker, Zone 56B
- Units 1 and 2 EDG Building, Switchgear Room Train B, Zone 56C
- Units 1 and 2 EDG Building, Diesel Generator 2C, Zone 57
- Units 1 and 2 EDG Building, Diesel Generator 1B, Zone 58
- Units 1 and 2 EDG Building, Diesel Generator 2B, Zone 59
- Units 1 and 2 EDG Building, Diesel Generator 1C, Zone 60
- Units 1 and 2 EDG Building, Diesel Generator 1-2A, Zone 61
- Units 1 and 2 EDG Building, Fire Protection Cardox Control Panel Hall, Zone 71

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

External Flooding Review. The inspectors reviewed plant design features that protect against external flooding and related licensee procedures to verify the licensee's flood mitigation plans and equipment were consistent with the design requirements and risk analysis assumptions. The inspectors reviewed flood protection barriers which included plant yard drains and the auxiliary and diesel generator building roofs. The inspectors also reviewed condition reports and maintenance work orders to verify the licensee was identifying and resolving problems. Documents reviewed are listed in the Attachment.

Internal Flooding Review. The inspectors walked down the following four areas to verify that plant design features and plant procedures for flood mitigation were consistent with the design requirements and internal flooding analysis assumptions. This included potential sources of internal flooding, the condition of room penetrations, and the condition of the sumps in the rooms. The inspectors also reviewed CRs and maintenance work orders to verify the licensee was identifying and resolving problems. Documents reviewed are listed in the Attachment.

- Unit 1 Auxiliary Feedwater (AFW) general area (Rooms 190 and 194)
- Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) pump room (Room 193)
- Unit 1 ‘A’ Motor Driven Auxiliary Feedwater (MDAFW) pump room (Room 191)
- Unit 1 ‘B’ MDAFW pump room (Room 192)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

Annual Review. The inspectors reviewed test results of FNP-0-ETP-4367, Performance Test for Unit 1 & 2 Colt-Pielstick (Large) Diesel Generator Jacket Water Heat Exchangers, for the 1B EDG to verify the licensee had adequately identified and resolved any potential heat exchanger deficiencies which could mask degraded performance, common cause heat sink performance problems that could increase risk, and heat sink performance problems that could result in initiating events or affect multiple heat exchangers in mitigating systems. The inspectors also reviewed UFSAR Section 9.5.5 and system design document A-181005, Diesel Generator System, to verify the acceptance criteria for FNP-0-ETP-4367 was appropriate. The inspectors walked down the 1B EDG room to verify the material condition of the heat exchangers was not degraded and that any existing deficiencies had been identified. The inspectors also reviewed the licensee’s CR database to verify that heat exchanger problems were being identified and resolved.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

Piping Systems ISI. 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. The inspectors selected a sample of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI required examinations and Code components in order of risk priority. The inspectors reviewed records and conducted independent inspections for 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 (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI, IWB-3000 or IWC-3000 acceptance standards.

Visual Testing (VT):
- APRI-4501 through 4504-1DM, Pressurizer Relief Nozzle and Three Pressurizer Safety Nozzles, ASME Class 1
- APR1-4205-49DM, Pressurizer Spray Nozzle, ASME Class 1

Ultrasonic Testing (UT):
- APR1-4205-49DM, Pressurizer Spray Nozzle, ASME Class 1
- APRI-1300-S02-S13, 16, 17 & 18, Reactor Pressure Vessel Studs, Class 1

Qualification and certification records for examiners, inspection equipment, and consumables along with the applicable NDE procedures for the above ISI examination activities were reviewed and compared to requirements stated in ASME Section V and Section XI.

The inspectors performed a review of an ISI related problem that was identified by the licensee and entered into the corrective action program. The inspectors reviewed the CR to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions.

Boric Acid Corrosion Control Program (BACCP) ISI. The inspectors reviewed the licensee's BACCP to ensure compliance with commitments made in response to NRC Generic Letter 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 reviewed records and walked down of parts of the reactor building that are not normally accessible during at-power operations to evaluate compliance with licensee BACCP requirements and verify that the boric acid visual examinations focused on locations where boric acid leaks can 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 licensee's procedures for implementation of the boric acid corrosion control program to ensure that they were in compliance with available industry guidance. The inspectors reviewed the following eight engineering evaluations completed for boric acid found on reactor coolant system piping and components to verify that the minimum design code required section thickness had been maintained for the affected component(s). The inspectors also reviewed licencee corrective actions implemented for evidence of boric acid leakage to confirm that they were consistent with requirements of Section XI of the ASME Code and 10 CFR 50, Appendix B, Criterion XVI.

- 2283, Pump Casing Drain Flange Leak
- 2286, 2B Boric Acid Transfer Pump Shaft Leak
- 2287, 2B Boric Acid Tank Manway Gasket
- 2289, 2A Boric Acid Transfer Pump Shaft Leak
- 2290, 2A Boric Acid Tank Manway Gasket
- 2291, 2A Boric Acid Tank Flow Orifice Gasket
- 2288, 2B Boric Acid Tank Flow Orifice Gasket
- 2292, Boric Acid to Blender Supply Valve

Documents reviewed are listed in the Attachment.

b. <u>Findings</u>

No findings of significance were identified.

1R11 <u>Licensed Operator Requalification</u>

a. <u>Inspection Scope</u>

<u>Quarterly Resident Review</u>. 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 Operator Continuing Training Program Administration. The inspectors observed scenarios conducted in the licensee's simulator for reactor startup and power ascension. 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.

<u>Annual Review of Licensee Requalification Examination Results</u>. On June 30, the licensee completed the requalification annual operating tests, required to be given to all licensed operators by 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the written examinations, individual operating tests, and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

Resident Quarterly Review. The inspectors reviewed the following two issues to verify implementation of licensee procedures FNP-0-87, Maintenance Rule (MR) Scoping Manual; NMP-ES-021, Structural Monitoring Program for the Maintenance Rule; and FNP-0-89, FNP Maintenance Rule Site Implementation Manual; and compliance with 10CFR50.65. The inspectors assessed the licensee's evaluation of appropriate work practices, common cause failures, functional failures, maintenance preventable functional failures, repetitive failures, availability and reliability monitoring, trending and condition monitoring, and system specialist involvement. The inspectors also interviewed maintenance personnel, system specialists, the MR coordinator, and operations personnel to assess their knowledge of the program.

- CR 2005111288, Unit 2 2C Service Water pump failure to start and equipment alignment
- CR 2005112472, Unit 1, 1F 4160V EPB meter and 1-2A Diesel Generator inoperability

Biennial Evaluation. The inspectors reviewed the licensee's most recent MR periodic assessment, Farley Nuclear Plant Units 1 and 2 Maintenance Rule Periodic Assessment, December 9-12, 2004, covering the 18 month period of December, 2002 through November, 2004. This report was issued to satisfy paragraph (a)(3) of 10 CFR 50.65, and covered the indicated period for the two units. The inspection was to determine the effectiveness of the assessment to verify it was issued in accordance with the time requirement of the MR, and to verify the evaluation included balancing reliability and unavailability, (a)(1) activities, (a)(2) activities, and use of industry operating experience. To verify compliance with 10 CFR 50.65, the inspectors reviewed selected MR activities covered by the assessment period for the following MR systems and equipment: Nuclear Instrumentation, Service Water, Radiation Monitors, Steam Generator Blowdown, Emergency Diesel Generators, and Reactor Makeup Water. Documents reviewed are listed in the Attachment.

The inspectors reviewed the site MR implementing procedures, relevant condition reports, plant major issues list, system health reports, and MR expert panel meetings minutes. In addition, the inspectors reviewed (a)(1) systems' monthly status, and MR monthly reports, Units 1 and 2 MR comprehensive unavailability reports, structural monitoring program 5-Year periodic inspection, and functional failures reports. The inspectors discussed issues with cognizant system engineers. Operational event information was evaluated by the inspectors in its use in MR functions. The inspectors selected system health reports and other corrective action documents of risk significant systems recently removed from 10 CFR 50.65 (a)(1) status and those in (a)(2) status for some period to assess the justification for their status. The documents were compared to the site's MR program criteria, and MR (a)(1) evaluations and scoping documents. The inspectors also reviewed corrective actions and acceptance criteria for systems in (a)(1) such as Radiation Monitors and Penetration Room Filtration systems to verify proper thresholds for entering systems into (a)(1) and timeliness commensurate with risk significance in resolving problems with the systems.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors assessed the licensee's planning and control for the following five planned activities to verify the requirements in licensee procedures FNP-0-ACP-52.3, Guidelines for Scheduling of On-Line Maintenance; NMP-GM-006, Work Management; and FNP-0-AP-16, Conduct of Operations - Operations Group; and the MR risk assessment guidance in 10 CFR 50.65(a)(4) were met.

- CR 2005110094, Load Center Busses 1-2K and 1-2L on alternate power sources.
- CR 2005109762, 1D Load Center Room Cooler Troubleshooting.
- CR 2005110111, Main Steam Isolation Valve Closing.
- CR 2005110485, Main Transformer Backfeed Alignment Problem.
- CR 2005111480, 1B EDG Jacket Water Leak.

b. Findings

Introduction. A Green, self-revealing non-cited violation (NCV) of 10 CFR 50.65(a)(4) was identified when the licensee failed to perform a required risk assessment following alignment of the 1-2K and 1-2L 600V load centers to their emergency power (Unit 1) supplies. This resulted in the risk being elevated from a Yellow to an Orange status without senior management concurrence and no additional compensatory actions.

Description. On October 14, the licensee aligned 1-2K and 1-2L to their emergency power supplies in preparation for upcoming train outages during the Unit 2 refueling outage. The loads on these busses were mainly supportive of the Service Water (SW) system and consisted of SW Booster Pumps, SW Battery Chargers, SW Cyclone Separator, and motor operated valves allowing SW Emergency Recirculation to the SW pond. This activity had been included in the risk assessment, but was moved ahead of the prescribed time; therefore, the risk assessment was not accurate for actual plant conditions.

Procedure FNP-0-ACP-52.3, Mode 1, 2, 3 Risk Assessment, required that the Shift Supervisor (SS) perform a risk assessment for all unscheduled activities. The "nights" SS attempted to update the risk assessment, but was unsuccessful. On October 15, the "days" SS successfully calculated an Orange risk condition which required Operations

Shift Superintendent and Operations Manager concurrence before authorizing work. Believing this to be in error, the SS performed multiple outage risk calculations but was unable to clear the Orange risk condition with 1-2K and 1-2L on their emergency power supplies. The SS then realigned 1-2K and 1-2L to their normal power supplies to clear the Orange risk condition.

Analysis. This finding is more than minor because it impacted the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the cornerstone objective in that the licensee failed to perform a risk assessment following a change in actual plant configuration and did not establish compensatory measures consistent with the elevated outage risk. This finding was evaluated using IMC 0609, Appendix K, Maintenance Risk Assessment Significance Determination Process, and based on an Incremental Core Damage Probability Deficit (ICDPD) of 1.03E-6 and no Risk Management Actions, this finding was initially assessed as greater than Green. However, the licensee's risk assessment model had not been updated to reflect a recent design change, which removed the SW booster pumps as the SW pump lubrication and cooling water source. Using a revised risk assessment model, which included this design change, the ICDPD was recalculated to be less than 1E-6, resulting in this finding being of very low safety significance (Green). This finding involved the cross cutting aspect of Human Performance in the area of Personnel for failure to follow procedure FNP-0-ACP-52.3.

Enforcement. 10 CFR 50.65(a)(4) requires that the licensee shall assess and manage the increase in risk that may result from proposed maintenance activities. Procedure FNP-0-ACP-52.3 implemented the requirements of 10 CFR 50.65 (a)(4) by requiring a risk assessment for all unscheduled activities. Contrary to the above, on October 14, 2005, a risk assessment was not performed after 600V load centers 1-2K and 1-2L were placed on their emergency power supplies. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as CR 2005110094, this violation is being treated as an NCV in accordance with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000364/2005005-01, Failure to Perform Risk Assessment.

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 and FNP-0-ACP-9.2, Operability Determination 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 Functional System Descriptions (FSDs) to verify system operability was not affected.

- CR 2005110060, 1D 600V Load Center room cooler trip
- CR 2005109569, 2B Centrifugal Charging Pump Damage
- CR 2005111036, Flow Accelerated Corrosion data of AFW/MFW tie

- CR 2004001493, Pull Box Inspection of Electrical Splices
- CR 2005111594, 1-2A EDG Syncroscope power supply

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.

- FNP-2-ETP-4460, Obtain Service Water Flows for A and B Containment Coolers
- FNP-2-STP-40.7, Emergency Core Cooling System (ECCS) Branch Line Flow Verification and Charging Pump Low Discharge Head Flow Test
- FNP-2-STP-40.0, Safety Injection With Loss of Off-Site Power Test
- FNP-1-STP-17.0B, Containment Cooling System Train B Operability Test
- FNP-2-STP-33.0B, Solid State Protection System Train B Operability 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 TS required specification. Shut-down 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 Cooling

b. Findings

Introduction. A Green, self-revealing NCV of Technical Specification (TS) 5.4.1.a was identified for inadequate procedural controls to maintain reactor coolant system (RCS) water level instruments operable. The lack of adequate procedural controls resulted in one of the required water level instruments being isolated prior to lowering RCS water level.

Description. On October 19, Unit 2 RCS water level was drained below the indicated pressurizer level in preparation for head removal. The drain down was conducted using procedure FNP-2-SOP-1.6, Draining The Reactor Coolant System. During the time the drain down occurred, the time to boil in the vessel was less than 15 minutes. Both trains of RHR were in service, but the steam generators were not available.

A temporary RCS level instrument was installed using FT-416 to provide an independent water level indication once the pressurizer was drained. The other independent water level indication came off a single tap of an RCS loop and provided indication on a tygon hose and LT-2965. Following shift turnover, control room operators observed that FT-416 was reading six feet below LT-2965. The licensee investigated this condition and found that the variable leg isolation valve was closed and the reference leg was disconnected. With the variable leg isolation valve closed, FT-416 would not have indicated any changes in RCS water level. This condition was not identified by control room operators during the drain down. The licensee determined that, prior to the drain down, control room operators had requested that instrumentation technicians vent LT-416 from the Pressurizer Relief Tank (PRT). This was accomplished by disconnecting the upper level connection from the PRT. However, due to miscommunications, the technician disconnected the upper connection and had also closed the variable leg isolation valve.

Analysis. This finding is greater than minor because it affected the procedure quality attribute and the Mitigating Systems Cornerstone objective, in that, procedural controls for operability of RCS level instrument were not appropriate to the circumstances. This finding was determined to be of very low safety significance because actual water level was at the level planned for the drain down. This finding involved the cross cutting aspect of Human Performance in the area of Organization in that procedural controls were not adequate.

Enforcement. TS 5.4.1.a requires that written procedures be implemented covering the activities listed in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. RG 1.33, Appendix A, 9. d. 4. includes procedures for draining the reactor coolant system. Contrary to the above, on October 19, the licensee failed to provide procedures appropriate to the circumstances, which resulted in required water level instrument being inoperable when lowering RCS water level. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as CR 2005110381, this violation is being treated as a NCV in accordance with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000364/2005005-02, Failure to Provide Adequate Procedural Controls.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed surveillance test procedures and either witnessed the test or reviewed test records for the following eight 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; and attended selected briefings to determine if procedure requirements were met.

Surveillance Tests
- FNP-2-STP-608.0, Main Steam Safety Valve Operational Test
- FNP-2-STP-29.2, Shutdown Margin Calculation
- FNP-2-STP-627, Local Leak Rate Testing of Containment Penetrations
- FNP-2-STP-34.2, Containment emergency core cooling system (ECCS) Sump Intake Inspection

In-Service Tests (ISTs)
- FNP-2-STP-10.3, Emergency Core Cooling Valves IST and Power Operated Relief Valve Block Valve Stroke Test

RCS Leak Detection
- FNP-1-STP-9.0, RCS Leakage

Containment Isolation Valves
- FNP-2-STP-627, Local Leak Rate Testing of Containment Penetrations, Pen 43
- FNP-2-STP-627, Local Leak Rate Testing of Containment Penetrations, Pen 44

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (TMs)

a. Inspection Scope

The inspectors reviewed the following two TMs and associated 10 CFR 50.59 screening criteria against the system design bases information and documentation and the licensee's temporary modifications 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 these TMs.

- Unit 1 and Unit 2 Plant Process Computers
- FNP-2-SOP-58.0 Appendix C, Installation of Jumpers for the Spent Fuel Pool Air Handling Unit Temperature Switch TSL-3906

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (21 samples)

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 and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection activities; and assessed occupational exposures to 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 LHRA and VHRA keys and for storage of irradiated material within the Unit 2 spent fuel pool (SFP) area were reviewed and discussed in detail. Reviewed Unit 2 refueling outage tasks included reactor vessel head disassembly, removal from containment and temporary storage activities; removal of a penetration sample from the old reactor vessel head; steam generator sludge lance tasks; fuel off-load; scaffolding activities; radioactive filter change-out; and radioactive waste (radwaste) handling and storage. The inspectors attended pre-job briefings and reviewed radiation work permit (RWP) details to assess communication of radiological control requirements to workers. Occupational worker adherence to selected RWPs and HPT proficiency in providing job coverage was evaluated through direct observations and interviews with licensee staff. Electronic dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results with a focus on reactor vessel head disassembly and removal. Worker exposure as measured by ED and by licensee evaluations of skin doses resulting from discrete radioactive particle or dispersed skin contamination events since January 1, 2005, and during the current refueling outage activities were reviewed and assessed independently. For HRA tasks involving potentially significant dose rate gradients, e.g., reactor vessel head penetration sampling, the inspectors evaluated the potential for use of dosimeter multi-badging to monitor worker exposure.

Postings for access to radiologically controlled areas and physical controls for the Unit 2 containment and for Unit 1 and Unit 2 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 Unit 2 containment equipment and work locations, Unit 2 drumming/storage room, selected Unit 1 and Unit 2 RAB locations, and the temporary old reactor head storage location adjacent to the Old Steam Generator Storage Facility (OSGSF). Results were compared to current licensee surveys and assessed against established postings and radiological controls.

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 areas associated with refueling outage activities, the inspectors evaluated the implementation and effectiveness of administrative and physical controls including air sampling, barrier integrity, engineering controls, and postings. Licensee identification and assessment of potential radionuclide intakes by workers between January 1, 2005 through November 10, 2005, were reviewed and discussed.

Radiation protection activities were evaluated against UFSAR, TS, and 10 Code of Federal Regulations (CFR) Parts 19 and 20 requirements. Specific assessment criteria included UFSAR Section 11, Radioactive Waste Management, and Section 12, Radiation Protection; 10 CFR 19.12; 10 CFR 20, Subpart B, Subpart C, Subpart F, Subpart G, Subpart H, and Subpart J; TS Sections 5.4, Procedures, and 5.7, High Radiation Area Controls; and approved procedures. Documents reviewed are listed in the Attachment.

<u>Problem Identification and Resolution</u>. 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 NMP-GM-GL-02, Corrective Action Program Details and Expectations Guideline, Version 2.0. Licensee CR documents associated with access controls, personnel monitoring instrumentation, and personnel contamination events were reviewed. Documents reviewed are listed in the Attachment.

b. <u>Findings</u>

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee records to verify the accuracy of reported Performance Indicator (PI) data for the periods listed below. To verify the accuracy of the reported PI elements, the reviewed data were assessed against guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Revision 3, and the NEI Frequently Asked Questions list.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed the PI results from October 2004 through September 2005. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and CRs related to controls for exposure significant areas. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in the Attachment.

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 corrective action program. This review was accomplished by reviewing daily hard copy summaries of CRs and by reviewing the licensee's electronic CR database.

.2 Annual Sample Review

a. Inspection Scope

The inspectors performed a detailed review of the history of issues resulting from water entering electrical pull boxes and damaging electrical cabling. In 2004, the 2E Service Water pump failed to start due to a corroded conductor that carries the signal to close the breaker and start the pump. CR 2004001493, AI 2004202088, and WO's S400424601 and S400424701 were reviewed to verify safety concerns were properly classified and prioritized for resolution; technical issues were evaluated and dispositioned to address operability and reportability; apparent cause determinations were sufficiently thorough; extent of condition, generic implications, common causes and previous history was adequately considered; and appropriate corrective actions (short and long-term) were implemented or planned in a manner consistent with safety and compliance. The inspectors also evaluated the documents against the requirements of the licensee's corrective action program as delineated in Procedure NMP-GM-003, "Corrective Action Program," and 10 CFR 50, Appendix B.

b. Findings and Observations

No findings of significance were identified. The inspectors reviewed FNP-0-GMP-60.1, General Inspection of Outdoor Electrical Duct Run Pull Boxes, and that a requirement to check all safety related pullboxes on a five year frequency had been included. Based on a review of the licensees responses to the sampled splices, discussion with the system engineer, and plant management, the inspectors concluded that the implementation of the corrective actions were thorough and adequate.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more safety significant safety issue. The inspectors' review focused on CRs with corrective action that were not sufficiently comprehensive to reduce the likelihood or prevent recurrence of the condition. The review also considered the results of the daily inspector CAP item screening discussed in Section 4OA2.1, licensee trending efforts, and licensee human performance results. The inspectors reviewed the licensee quarterly trend reports for May 2005 - July 2005, and August 2005 - October 2005, daily CRs, selected completed CRs, Maintenance Rule (a)(1) list, equipment health reports, and quality assurance reports to identify issues not recognized by the licensee. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. The inspectors also evaluated the reports against the requirements of the licensee's CAP procedures FNP-0-AP-30.0, Corrective Action Reporting and NMP-GM-002-GL05, Corrective Action Program Trend Coding and Analysis Guideline, and the requirements of 10 CFR 50, Appendix B.

b. Assessments and Observations

No findings of significance were identified. The inspectors noted that the licensee had identified adverse trends with mispositioned components and equipment clock resets. These were the two most significant issues noted by the inspectors during the semi-annual review period. These were continuing into the next quarter (November 2005, December 2005) and had not been analyzed by the licensee as their trend quarter had not ended.

4OA3 Event Follow-up

.1 Unit 2 Alert Declaration

a. Inspection Scope

On November 8, at 1:19 p.m, the licensee declared an Alert due to loss of the Unit 2 Control Board Annunciators for approximately 14 minutes. At the time of the event, the Unit 2 reactor was defueled. Control Board indication was restored and the Alert condition was subsequently exited at 1:49 p.m. The inspectors reviewed CR 2005111326 for this event.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 Unit 2 Reactor Pressure Vessel Head (RPVH) Replacement

a. Inspection Scope

RPVH and Control Rod Drive Mechanism (CRDM) Housing Fabrication. The inspectors observed/reviewed the activities detailed below to verify compliance with applicable construction and inspection Codes (ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda and Section XI, 1989 Edition as defined in DCP-03-2-9915 (2039991501), Replacement Reactor Vessel Head and Implement Simplified Head Assembly Upgrade. The inspectors reviewed the following records associated with the fabrication of the replacement RPVH:

- Certified Material Test Report (CMTR) for the RPVH forging (Material: SA-508, Grade 3, Class 1, Heat No. 03W108-1-1), including Magnetic Particle (MT) and UT examinations reports of the RPVH forging prior to welding

- CMTR for eight (8) CRDM to RPVH Adapters (Material: SB-167, N06690, Heat/Lot No. D272007), including Dye Penetrant Test (PT) and UT examinations, hydrostatic testing, and dimensional examinations reports

For the following five welds, the inspectors reviewed Fabrication Process sheets, Welding Work Record sheets, Fit-up Inspection records, PT examination records, RT (Radiograph) examination records, and UT examination records, as applicable:

- Latch Housing to Head Adapter Weld #s: WC-H202-1A, WC-H202-18A
- Rod Travel Housing and Latch Weld #s: WC-H009-15A, WC-H009-18A
- Cladding Weld #s: WO-A103-1, WO-A103-2
- J-Groove Buttering Weld #s: WO-A107-1A, WO-A107-29A

- J-Groove Weld #s: WC-A109-2, WC-A109-42A

In addition, the inspectors reviewed the following records associated with these listed welds:

- Welder qualification records for 17 welders

- Welding material certification records for: (1) Electroslag Cladding Material - ER309L Wire Heat/Lot No. NA547 and Flux PFB-7FK Lot No. 3H6812025 and ER308L Wire Heat/Lot No. 26547 and Flux PFB-7FK Lot No. 3H6812025, (2) E308L-16 - Heat/Lot A3501N, (3) E309-16 - Heat/Lot A1071216N, (4) ERNiCrFe-7 - Heat/Lot Nos. Nx3609JK qnd Nx3649JK, (5) ER316L - Heat/Lot Nos. BHA9400 and BF36099, and (6) ENiCrFe-7 - Heat/Lot Nos. 401393 and 312392

- Certification records for PT examination materials

- Nondestructive Examination (NDE) and General Inspection Personnel Certification records for 11 Level II PT examiners

- RT films and NDE examiner certification records for CRDM Rod Travel Housing to Latch Housing welds WC-H009-18A,WC-H009-4A ,WC-H009-15A and WC-H009-11A, and CRDM Latch Housing to RPVH Adapter welds WC-H202-1A and WC-H202-18A

- Non Conformance Reports (NCRs) UGNR-SFN2-RVH-002, Cladding Error, and UGNR-SFN2-RVH-004, PT Indications on J-Groove Weld

- Hydrostatic Test Record, including the chemical analysis report of the test water and reports of post hydrostatic test NDEs for the RPVH

- Final post weld heat treatment (PWHT) records, including time-temperature strip charts, Heat Treatment Operation Cards, and PWHT List, for the RPVH

- Westinghouse Calculation WB-CN-ENG-04-3 Farley CRDM- ASME Code Section XI Reconciliation, Revision 2

- Westinghouse Calculation CN-RCDA-04-8, Farley Units 1 and 2 RRVCH - ASME Section XI Reconciliation

Preservice Inspection (PSI) and Baseline Inspections. The inspectors reviewed selected NDE records, which documented the ASME Section XI PSI and baseline inspections performed to provide baseline conditions for future inspections in accordance with NRC Order EA-03-09. Relative to ASME Section XI PSI of the replacement RPVH, the inspectors reviewed the following completed PT records.

- 16 peripheral Category B-O CRDM Latch Housing to Rod Travel Housing welds
- 16 peripheral Category B-O CRDM Latch Housing to RPVH Adapter welds

- 4 peripheral Category B-O Instrumentation Port Head Adapter Flange welds
- 2 peripheral Category B-O Reactor Vessel Level Indication System (RVLIS) Head Adapter Flange welds

In addition, the inspectors reviewed: (1) personnel certification records for 2 Level III and three 3 Level II NDE Examiners, and (2) PT material certification records for the welds listed above.

In order to support future inspections required by NRC Order EA-03-09, the inspectors reviewed the scope of the licensee’s baseline inspections which consisted of:

- Automated open bore inside diameter UT and eddy current (ET) examination of the CRDM, RVLIS, and instrumentation penetrations covering from the bottom of the penetrations to a minimum of 2" above the J-groove welds

- ET examination of the J-groove surface for CRDM, RVLIS, Instrumentation Port, and vent nozzle penetrations

- ET examination of the outside diameter of the penetrations below the J-groove welds for all penetrations

- Under head PT examination of all penetration to head J-groove welds using “PT white” acceptance criteria (performed to obtain the best surface possible to prevent future primary stress corrosion cracking)

- Bare Metal VT (Visual) examination of the top of the RPVH

- Under-head VT examination of the RPVH

The inspectors reviewed a sample of the baseline NDE data consisting of :

- Completed UT and ET reports for penetration Nos. 5, 16, 42 and 54

- Personnel qualifications for UT, ET, and VT examiners performing baseline inspections

- Certification records of ET probe Serial No. 431048 and UT transducer Serial Nos. TX10021, TX10023, RX10023, RX10025, S8464, and 10026

- Under head PT examination reports of all penetration to head J-groove welds using “PT white” acceptance criteria, including personnel qualifications and PT material certifications

- Bare Metal VT examination records for VT examination of the top of the RPVH, specifically penetration Nos. 12, 2, 23, 45, and 32

Review of 10 CFR 50.59 Screening/Evaluation. The inspectors reviewed DCP 03-2-9915 (2039991501), Replacement Reactor Vessel Head and Implement Simplified Head Assembly Upgrade, including the associated 10 CFR 50.59 screening to verify that changes between the original RPVH and the replacement RPVH, and modifications resulting from installation of the replacement RPVH were properly evaluated in accordance with 10 CFR 50.59. This included verification that any change in weight between the old RPVH and the replacement RPVH had been taken into account in design stress calculations and rating of the polar crane.

Review of Quality Assurance (QA) Activities. The inspectors reviewed surveillance and audit reports of QA activities at the vendor facilities to verify that the licensee, or their representative, evaluated the QA activities performed by the fabricator (Mitsubishi Heavy Industries, (MHI)) and the contractor for RPVH fabrication (Westinghouse) at the MHI facilities. The inspectors verified that the QA audits and surveillances included activities such as: (1) review of fabricator procedures and fabrication records, (2) witnessing NDE examinations, material testing, and manufacturing processes, and (3) assessment of Westinghouse QA audits and surveillances of MHI.

Review of Radiation Protection Activities. The inspectors reviewed and evaluated radiation protection planning, preparation, and establishment of radiological controls to verify they were properly implemented. The inspectors reviewed, discussed, and evaluated As Low As Reasonably Achievable (ALARA) planning documents; initial dose estimates and actual results; exposure controls including temporary shielding, contamination and airborne radioactivity controls, and radioactive material management; and transportation work plan documents and emergency contingencies.

ALARA work plan details and recent dose reduction initiatives were reviewed. The radiation, contamination, and airborne radioactivity surveys in the packages were reviewed for radiological work conditions and the adequacy of prescribed postings and surveys. The inspectors reviewed RWPs to determine projected exposure, expected conditions, ED dose and dose-rate alarm settings, dosimetry requirements, protective clothing/equipment, worker instructions and HPT instructions. The ALARA exposure estimates were reviewed and evaluated against work scope/radiological conditions. Corrective action documentation was reviewed for significant trends or recurring problems with work practices and controls. The expected and measured source terms, and implementation of the Unit 1 reactor head replacement ALARA lessons-learned were discussed and evaluated.

The inspectors discussed the movement of the reactor vessel head from containment to the temporary storage facility. Dose-rate data and resultant cumulative dose to workers involved with initial transport and packaging of the old vessel head were reviewed and discussed in detail. The inspectors toured and conducted independent surveys of the temporary storage facility environs; and discussed radiological controls and temporary monitoring programs at the temporary storage area located adjacent to the OSGSF.

Licensee program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J, and approved licensee procedures.

Additional documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Independent Spent Fuel Storage Installation (ISFSI) Radiological Controls

a. Inspection Scope

The inspectors observed gamma-ray, neutron, and contamination surveys of the ISFSI facility and compared the results to previous surveys and TS limits. The inspectors also observed and evaluated implementation of radiological controls, including labeling and posting, and discussed the controls with a Health Physics (HP) Technician and HP supervisory staff. The HP work plan for cask loading and transport was also reviewed and discussed

Radiological control activities for ISFSI areas were evaluated against 10 CFR Part 20, 10 CFR Part 72, and Certificate of Compliance TS. Documents reviewed are listed the Attachment.

b. Findings

No findings of significance were identified.

.3 (Closed) TI 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) (Unit 2)

a. Inspection Scope

The inspectors reviewed the licensee’s final response to NRC Bulletin 2004-01, dated July 26, 2004. The inspectors verified that the licensee’s inspection activities conducted during this outage were consistent with their response.

The inspectors conducted an independent observation of the susceptible welds on the top of the pressurizer to ensure that the physical conditions of the pressurizer penetrations and welds were adequately inspected, and that there were no problems with debris, insulation, dirt, boron from other sources, physical layout, or viewing obstructions, which could have interfered with the identification of boric acid leakage. Specifically, the inspectors observed three safety valve nozzles, the common relief valve nozzle, and the spray line. The inspectors also reviewed licensee records of the inspection of these nozzles.

b. Findings

No findings of significance were identified. Reporting Requirements are as follows:

a. For each of the examination methods used during the outage, was the examination:

1. Performed by qualified and knowledgeable personnel?

   Yes. The licensee used a knowledgeable staff member certified as Level II, VT-2 examiner. The qualification and certification procedure referenced the industry standard ANSI/ANST CP-189, Standard for Qualification and Certification of Nondestructive Testing Personnel.

2. Performed in accordance with demonstrated procedures?

   Yes. The licensee performing the bare metal inspection of the pressurizer penetrations in accordance with procedure FNP-0-NDE-100.22, Visual Examination VT-2, Version 5.0.

3. Able to identify, disposition, and resolve deficiencies?

   Yes. The inspectors concluded that the licensee's direct visual examinations were capable of detecting leakage from cracking in pressurizer penetrations if it had existed. This conclusion was based upon the inspectors' direct observations of pressurizer penetration locations, which were free of debris or deposits that could mask evidence of leakage in the areas examined. The inspectors also verified that the licensee's procedures included guidance for proper disposition and investigation of any identified deficiencies.

4. Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01?

   The inspectors verified that the licensee's examination personnel were capable of identifying any leakage in pressurizer penetration nozzles.

b. What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system?

   Through observations, the inspectors verified that the metal reflective insulation had been removed. The licensee personnel performed a direct visual inspection of these pressurizer penetrations. Based on this examination, the area examined was generally clean and free of debris or deposits or other obstructions which could mask evidence of leakage.

c. How was the visual inspection conducted?

   The licensee's inspection personnel used the direct visual examination technique.

d. How complete was the coverage?

The licensee was able to view the entire circumference, 360°, around each penetration.

e. Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized?

The examination personnel were appropriately trained and qualified to identify small boron deposits as described in the bulletin.

f. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

There were no deficiencies identified that required repair.

g. What, if any, impediments to effective examinations, for each of the applied methods, were identified?

There were no impediments for an effective examination.

h. If volumetric or surface examination techniques were used for the augmented inspections, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations?

Not Applicable. No augmented surface or volumetric examinations were performed. In accordance with the licensee's response, only a Bottom Mounted Vessel (BMV) examination was conducted during this outage, and there were no indications identified that required further examination.

I. Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system?

Not Applicable. There were no indications of boric acid leaks from susceptible pressure-retaining components.

4OA6 Meetings, Including Exit

On January 5, 2006, 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 was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

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

NRC personnel
M. Widmann, Chief, Reactor Projects, Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000364/2005005-01	NCV	Failure to Perform Risk Assessment (Section 1R13)
05000364/2005005-02	NCV	Failure to Provide Procedural Guidance (Section 1R20)

Closed

2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment
Complete Walk-Down
UFSAR Section 6.3
Technical Specification Section 3.5.2
Student Lesson Plan OPS-52102B, OPS-40302C
ECCS Function System Description

Section 1R06: Flood Protection Measures
External Flooding
Drawings: D-171488, D-171426, D-175209, and D-175005
FNP-0-AOP-21.0
UFSAR Sections 2.4.2 and 3.4

Internal Flooding
Calculation Number BM-99-1932-001
REA 00-2354

Section 1R08: Inservice Inspection Activities
FNP-0-ETP-4496, Corrosion Assessment, Version 2.0
NMP-ES-019, Boric Acid Corrosion Control Program, Version 1.0
NMP-ES-019-GL01, Boric Acid Corrosion Control Implementation Guideline, Version 1.0
NMP-ES-019-GL02, Boric Acid Corrosion Control Health Report Guideline, Version 2.0
NMP-ES-019-GL03, Boric Acid Deposit Sampling, Analysis and Data Evaluation, Version 2.0
FNP-0-101, Boric Acid Corrosion Control Program, Version 10
Condition Report 2005110494, Use of Wrong Cable Length for Ultrasonic Testing
Boric Acid Corrosion Health Report, 3rd quarter, 2005

Section 1R11: Licensed Operator Requalification
FNP-2-STP-29.6, Calculation of Estimated Critical Condition
FNP-2-UOP-1.2, Startup of Unit From Hot Standby to Minimum Load
FNP-2-UOP-3.1, Power Operation

Section 1R12: Maintenance Effectiveness
Biennial Review
Administrative Procedures
FNP-0-M-87, Maintenance Rule Scoping Manual, Rev. 15
FNP-0-M-89, Maintenance Rule Site Implementation Manual
FNP-0-SYP-17, Maintenance Rule Monitoring and Reporting
FNP-0-SYP-19, Performance Criteria for Systems Under the Scope of the Maintenance Rule, Rev. 6

System Health Reports
Emergency Diesel Generator, 3rd Qtr 2005
Nuclear Instruments, 2nd Qtr 2005
Radiation Monitors, 3rd Qtr 2005
Reactor Makeup Water System, 2nd Qtr 2005
Service Water, 3rd Qtr 2005
Steam Generator Blow Down, 1st, 2nd, & 3rd Qtrs 2005

Condition Reports
2002001474, 2003001749, 2003001937, 2003002210, 2003002541, 2003002578, 2003002598, 2004000408, 2004001637, 2004002380, 2004100183, 2004101110, 2004101906, 2004102139, 2005101342, 2005101343, 2005111977, 2005112217, 2005112220

Maintenance Rule Expert Panel Meetings
Meeting #51, 08/10/04
Meeting #52, 09/23/04
Meeting #55, 01/28/05
Meeting #61, 06/28/05
Meeting #64, 10/06/05

Maintenance Rule Reports
A1 SSC Monthly Status Report, Nuclear Instrumentation System, 01/01, 06/01, 02/02, 10/05
A1 SSC Monthly Status Report, Gammametrics System, 10/03
A1 SSC Monthly Status Report, Unit 2 SW Lube & Cooling, 08/05
A1 SSC Monthly Status Report, Diesel Generator Annunciators, 10/05
A1 SSC Monthly Status Report, Unit 2 Steam Generator Blow Down, 08/05
A1 SSC Monthly Status Report, Radiation Monitors R11/12, 10/05
Maintenance Rule Monthly Report, 08/05, 09/05

Audits and Self Assessments
Farley Maintenance Rule Periodic Assessment, December 2004

Miscellaneous
Structural Monitoring Program, 1st 5-Year Periodic Inspection Period, 12/2001
Structural Monitoring Program, Structural Inspection Database, 1996
Unit 1 & 2 Comprehensive Unavailability Report, 2000-2005
Unit 1 & 2 Availability Report, 10/2002-10/2005
Functional Failures Report, 11/2000-11/2005
Farley Major Issues List, Rev. 10/18/05

Section 1R20: Refueling and 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 Q2B21FT0416
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
Westinghouse Low Power Physics Tests results for Farley Unit 2 Cycle 18

Section 2OS1: Access Controls to Radiologically Significant Areas

Procedures/Guidance Documents

Farley Nuclear Plant (FNP) Administrative Procedure (AP)-90, ALARA Policy and Implementation, Version (Ver.) 4.0

FNP Radiation Control and Protection Procedure (RCP)-0.1, Key Control Program and Health Physics Guidance for Control of High Radiation Areas, Radiological Exclusion Areas (Locked High Radiation Areas), and Very High Radiation Areas, Ver. 9.0

FNP-0-RCP-4, Refueling Survey, Ver. 18.0

FNP-0-RCP-13.1, Use of the HIS-20 RWP Section, Ver. 15

FNP-0-RCP-26.0, Radiological Surveys and Monitoring, Ver. 32

FNP-0-RCP-190, Skin Dose Assessment Due to Contamination on Personnel Skin or Clothing,

FNP-0-Dosimetry Procedure (DOS) -1, Personnel Monitoring, Ver. 43

Nuclear Management Procedure-GM-002-GL02, Corrective Action Program Details and Expectations Guidelines, Ver. 6

Heath Physics Plan for the Shipment of the Old Reactor Head Packages, 10/12/05

Unit 2 Refueling Outage 17 (2R17) Health Physics Reactor Head Replacement Work Plan, 10/12/05

2R17 Health Physics Old Reactor Head Transportation Plan, 10/12/05

RWP 05-0451, Work Associated with Inspections, Observations, Minor Maintenance in Posted Exclusion/Locked High Rad Areas Including the Change Out and Transport of Mechanical Filters > 1 Rem /Hr @ 30 cm, Revision (Rev.) 0

RWP 05-2301, All Work Associated with General Health Physics Coverage in the Auxiliary Building, or Unit 2 (U2) Containment to Support the 2R17 Outage, Rev. 0

RWP 05-2461, All Work Associated with Disassembly Reassembly of the Reactor (Rx) Head in Unit 2 Containment to Support the 2R17 Outage, Rev. 0

RWP 05-2480, All Work Associated with Routine Scaffolding Activities in the Auxiliary (Aux) Building and the Unit 2 (U2) Containment to Support the 2 R17 Outage, Rev. 0

RWP 05-2623, All Work Associated with Removal of a Penetration Sample from Under the Reactor Head in the Unit 2 (U2) Containment, Rev. 1

RWP 05-2625, All Low Radiological Conditions Work Associated with Replacement of the U2 Rx Head (No LHRA Allowed), Rev. 1

RWP 05-2626, All Work Associated with Moderate Radiological Conditions and Locked High Rad Activities Associated with Replacement of the U2 Rx Head, Rev. 0

RWP 05-2734, All Work Associated with Sludge Lance and FOSAR Inspsections of Steam Generator Secondary Sides to Support the 2R17 Outage, Rev. 0

Data/Records

Assessment of Performance for RWP 04-1626, Reactor Head Replacement

Unit 2 Spent Fuel Pool Non-Fuel Items Inventory Listing, as of 11/03/05

Worker Deep Dose Equivalent Data for RWP 05-0451 Radiation Controlled Area (RCA) Entries 11/07/05

Dose Report Data: Task Accumulative Dose, Maximum Dose, and Maximum Dose Rates; and Individual Worker Accumulative Dose, Maximum Dose, and Maximum Dose Rates, as of 11/10/05 for RWP 05-2623, RWP 05-2734, RWP 05-2625, RWP 05-2626,

Old Reactor Head Contamination Survey and Inspection Data: Survey Number (No.) 20157, Date Conducted (11/05/05); No 18946, (10/08/05); No. 18431, (09/10/05); No. 17927, (08/13/05); No. 17497, (07/16/05); No. 17008, (06/14/05); No. 16570, (05/13/05); No. 16171, (04/13/05); No. 15734, (03/13/05); No. 15377, (02/13/05)

HP Survey Data of Outside Radioactive Material Laydown Area (HP204) Including Boundary of

Old Reactor Head Storage Area: Survey Number (No.) 20158, Date Conducted (11/05/05); No. 18961, (10/09/05); No. 18432, (09/10/05); No. 17926, (08/13/05); No. 17577, (07/23/05); No. 17477, (07/16/05; No. 17164, (06/25/05); No. 16570, (05/28/05); No. 16389, (04/30/05); No. 16017, (04/02/05); No.15646, (03/06/05); No. 15295, (02/05/05); No 14896, (01/08/05); No. 14549, (12/15/04); No. 14059, (11/14/04)
Perimeter Monitoring Thermoluminescent Dosimeter (TLD) Location Data: 1st Half CY 2005
FNP-0-RCP-4 Appendix A, Record of Dose Rates During Initial Fuel Assembly Transfer, conducted 10/25/05
HP Form 118, HP Locked High Radiation Area Control Log, 11/07-09/05
Radiological Survey No. 20252, Tri-Nuclear Filters in U2 Spent Fuel Pool, 11/7/05
Negative Pressure Unit/HEPA I.D. No. HP-NPU-031, In-Use Ventilation Unit Inspection Sheet, 11/9/05

Corrective Action Program (CAP) Documents
Review of Locked High Radiation Area (LHRA) Administrative Controls, 08/9-12/05
EV-04-2156, Southern Nuclear Company Fleet High Radiation Area Control Self-Assessment, 11/01/04
Quality Assurance Surveillance Number 2004-004, Primary Spent Resin Cask Shipment, 02/17-18/04
Health Physics Assessment: Inefficient Performance of HP Routine Activities, 06/14-18/04
CRs: 2004102656, 2004104039, 2004104040, 2005110317, 2005110491, 2005110574, 2005111241

Section 4OA1: Performance Indicator Verification
Procedures/Guidance Documents
FNP-0-AP-54, Preparation and Reporting of NRC Performance Indicator Data and NRC Operating Data, ver. 6.0

Data/Records
Electronic dosimeter dose rate alarm logs, November 2004 - October 2005

CAP Documents
CR No. 2005107891, Tri-nuclear Vacuum Not Adequately Secured in Spent Fuel Pool, 08/05/05
CR No. 2005108276, Spike in Dose Rate Detected During Radiography Testing, 08/18/05
CR No. 2005106964, Spike in Dose Rate Detected During Spent Fuel Dry Storage Cask Loading, 07/13/05

Section 4OA5: Other
4OA5.1
Westinghouse Project Plan PP-MSI-02-9, Farley Units 1 and 2 Replacement Reactor Vessel Head (RRVH) & Head Assembly Upgrade Package (HAUP)
Westinghouse Supplier Quality Program Audit Report WES-2004-190 dated October 4, 2004
Mitsubishi Heavy Industries Ltd. (MHI) Manufacturing and Inspection Plan UGS-L5-020113, Farley Units 1 & 2 Replacement Reactor Vessel Closure Head Project
Westinghouse Programs & Supplier Qualification Audit 18701 - Section IV - Procurement
Nuclear Procurement Issues Committee Audit Checklist - Section IV - Procurement - Audit QAA-05-032/NUPIC 19323
Nuclear Procurement Issues Committee Audit Checklist - Section IV - Procurement - Audit QAA-05-032

Westinghouse Surveillance Plan/Report RRVCH-SR-FAR2-030, Farley RVCHR - Final Acceptance and Release
Westinghouse Surveillance Plan/Report RRVCH-SR-FAR2-024, Liquid Penetrant Examination
Farley Unit 2 Replacement Rx Vessel Closure Head Pre-Service Inspection Final NDE Report WDI-PSI-1302843-FSR-001
Design Change Package DCP-03-2-9915 (2039991501), Replacement Reactor Vessel Head and Implement Simplified Head Assembly Upgrade

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ISFSI Procedures/Guidance Documents
FNP-0-STP-822, HI-STORM Overpack Surface Dose Rates, Ver. 2.0
Health Physics Dry Fuel Storage Work Plan, Rev. 2B
Certificate of Compliance No. 1014, Holtec HI-STORM 100 Cask System, Amendment 1, Appendix A, Technical Specifications

ISFSI Data/Records
Radiological Survey No. 18151, ISFSI Area, 8/25/05