UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

April 30, 2010

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2010002, 05000260/2010002 AND 05000296/2010002

Dear Mr. Krich:

On March 31, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. The enclosed inspection report documents the inspection results which were discussed, on April 1 and 9, 2010, with Mr. James Randich 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.

In addition to the routine Reactor Oversight Process baseline inspections for all three units, the inspectors continued to apply the Augmented Inspection Plan on Unit 1 as delineated in NRC letters dated May 16, 2007, December 6, 2007 and May 21, 2008. This Unit 1 Augmented Inspection Plan was conducted to compensate for the lack of valid data for certain Performance Indicators (PI). These additional inspections were only considered to be an interim substitute for the invalid Unit 1 PIs until complete and accurate PI data was developed and declared valid. However, subsequent to the Unit 1 startup on May 22, 2007, the PIs in the Initiating Events and Barrier Integrity cornerstones, and the Safety System Functional Failure PI of the Mitigating Systems cornerstone, have become valid as acknowledged by the Tennessee Valley Authority letters dated January 7, 2008 and July 11, 2008. Consequently, the only PIs that remain invalid, and thereby subject to the augmented baseline inspection, were the Mitigating Systems Performance Index PIs.

This report documents five NRC-identified findings of very low safety significance (Green). These five findings were also determined to involve violations of NRC requirements. Additionally, three licensee-identified violations 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 they were entered into your corrective action program, the NRC is treating all of these findings as noncited violations (NCVs) consistent with Section VI.A.1 of the NRC 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 U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at the Browns Ferry 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 Senior Resident Inspector at the Browns Ferry Nuclear Plant. The information you provide will be considered in accordance with 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 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/

Eugene F. Guthrie, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-259, 50-260, 50-296
License Nos.: DPR-33, DPR-52, DPR-68

Enclosure: Inspection Report 05000259/2010002, 05000260/2010002 and 05000296/2010002
w/Attachment: Supplemental Information

cc w/encl. (See page 3)

this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at the Browns Ferry 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 Senior Resident Inspector at the Browns Ferry Nuclear Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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ADAMS: ☐ Yes ACCESSION NUMBER:_______________________ ☐ SUNSI REVIEW COMPLETE

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DATE	04/29/2010	04/29/2010	04/29/2010	04/28/2010	04/30/2010	
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Letter to R. M. Krich from Eugene F. Guthrie dated April 30, 2010

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2010002, 05000260/2010002 AND 05000296/2010002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	050002592010002, 05000260/2010002 and 05000296/2010002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	January 1, 2010 through March 31, 2010
Inspectors:	T. Ross, Senior Resident Inspector C. Stancil, Resident Inspector K. Korth, Resident Inspector R. Baldwin, Senior Operations Engineer (1R11.2) M. Coursey, Reactor Inspector (1R08) H. Gepford, Senior Health Physicist (2RS1, 4OA1.2)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000259/2010002, 05000260/2010002 and 05000296/2010002; 01/01/2010 – 03/31/2010; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Maintenance Effectiveness, Refueling and Other Outage Activities, and Identification and Resolution of Problems.

The report covered a three month period of inspection by resident inspectors and reactor inspectors from the region. Five noncited violations (NCV) were identified. The significance of most findings is identified 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: Initiating Events

Green. The inspectors identified a noncited violation of Technical Specifications (TS) 5.4.1.a for failure to follow surveillance procedure 3-SR-3.4.9.1(2), Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring during In-service Hydrostatic Leak Testing, to ensure all required Unit 3 temperatures were being monitored and verified to meet TS 3.4.9, RCS Pressure and Temperature Limits. Unit 3 reactor operators selected a wrong reactor pressure vessel (RPV) metal temperature to monitor, and the operator and Unit Supervisor (US) failed to recognize that the incorrect RPV temperature being monitored was outside the TS 3.4.9 limits. The licensee subsequently verified all required RPV temperatures were within TS 3.4.9 limits. This issue was entered into the licensee's corrective action program as problem evaluation report (PER) 222844.

This finding was determined to be of greater than minor significance because it was associated with the Initiating Events Cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. More specifically, the lack of reactor operator attention, and US oversight, during the RPV in-service leak test, resulted in operator errors that adversely affected the operators' ability to monitor and verify RPV metal temperatures were within TS Figure 3.4.9-2 limits to preclude a low temperature overpressure event. The finding was determined to be of very low safety significance according to Inspection Manual Chapter 609.04, Phase 1 - Initial Screening and Characterization of Findings, because it did not actually exceed the TS limit or adversely affect any mitigating systems. The cause of this finding was directly related to the cross-cutting aspect of Human Performance and Error Prevention in the Work Practices component of the Human Performance area, because human performance errors by the control room operators resulted in selecting the wrong RPV metal temperature to monitor and not recognizing this temperature exceeded TS limits [H.4.(a)]. (Section 1R20.1.2)

Green. The inspectors identified a noncited violation of Technical Specifications (TS) 5.4.1.a for failure to establish an adequate surveillance procedure to ensure all relevant reactor pressure vessel (RPV) metal temperatures of all four RPV regions were being monitored during the Unit 3 RPV in-service leak test pursuant with TS Surveillance Requirement (SR) 3.4.9.1, RCS Pressure and Temperature Limits. The licensee subsequently verified all required RPV temperatures were within TS 3.4.9 limits. This issue was entered into the licensee's corrective action program as PERs 223539 and 224778.

This finding was determined to be of greater than minor significance because it was associated with the Initiating Events Cornerstone attribute of Procedure Quality, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. More specifically, the procedure used by operators to monitor RCS and RPV temperatures, during the RPV in-service leak test, lacked sufficient details to ensure all relevant RPV temperatures would be monitored to meet TS SR 3.4.9.1 which could increase the likelihood of a low temperature overpressure event. The finding was determined to be of very low safety significance according to Inspection Manual Chapter 0609, Phase I - Initial Screening and Characterization of Findings, because it did not actually exceed the TS limit or adversely affect any mitigating systems. The cause of this finding was directly related to the cross-cutting aspect of Complete and Accurate Procedures in the Resources component of the Human Performance area because the applicable surveillance procedure lacked sufficient details and guidance to ensure all relevant RPV metal temperatures would be monitored pursuant to TS SR 3.4.9.1 [H.2.(c)]. (Section 1R20.1.3)

Cornerstone: Mitigating Systems

Green. The inspectors identified a noncited violation of 10 CFR 50.65(a)(2) for failure to demonstrate that the performance of the A3 Emergency Equipment Cooling Water (EECW) pump was effectively controlled by preventive maintenance (PM) such that the pump remained capable of performing its intended function. Also due to inadequate evaluations performed after the A3 EECW pump exceeded its Maintenance Rule a(2) performance criteria, goal setting and monitoring were not established as required by paragraph a(1) of the Maintenance Rule. The licensee subsequently declared the EECW system in (a)(1) status and was in the process of developing the required goals and monitoring plan. This issue was entered into the licensee's corrective action program as problem evaluation report 223404.

The finding was determined to be of greater than minor significance because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective of ensuring availability and reliability of systems designed to respond to initiating events to prevent undesirable consequences. More specifically, the licensee failed to demonstrate effective control of EECW system availability through appropriate PM. According to NRC Inspection Manual Chapter 0609.04, Phase I - Initial Screening and Characterization of Findings, this finding was determined to be of very low safety significance because it did not lead to an actual loss of a system safety function or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of this finding was directly related to the cross cutting aspect of Thorough Evaluation of Identified Problems in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee did not adequately evaluate the causes of the A3 EECW pump unavailability and thereby failed to correctly determine the impact on the 10 CFR 50.65(a)(2) unavailability performance criteria [P.1(c)]. (Section 1R12)

Green. The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to promptly recognize, and then correct in a timely manner, non-conforming conditions involving the in-service testing (IST) requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance (OM) of Nuclear Power Plants for the Equipment Cooling Water (EECW) system identified in June 2009. These nonconforming conditions involved the use of flow instrumentation without the proper accuracy, and failure to use the pre-service pump curve when establishing additional IST baseline reference values. The licensee revised the timeliness of their corrective action plans and decided to track this issue as a nonconforming condition. This issue was entered into the licensee's corrective action program as PER 225844.

The finding was determined to be of greater than minor significance because if left uncorrected it could become a more significant safety concern. In-service testing of the EECW system in conformance with the ASME OM Code provides assurance that degraded pump performance would be promptly detected and corrected. Failing to recognize and resolve these and other IST program deficiencies could lead to untimely detection of EECW pump degradation. According to Inspection Manual Chapter 0609.04, Phase I - Initial Screening and Characterization of Findings, this finding was determined to be of very low safety significance because it did not lead to an actual loss of a system safety function or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of this finding was directly related to the cross-cutting aspect of Appropriate and Timely Corrective Actions in the Corrective Action Program component of the Problem Identification and Resolution area because the licensee failed to take appropriate corrective actions to restore full compliance with the ASME OM Code requirements in a timely manner [P.1(d)]. (Section 4OA2.2)

Cornerstone: Barrier Integrity

Green. The inspectors identified a noncited violation of Technical Specifications 5.4.1.a for the failure to comply with operating procedures for Unit 3 new fuel receipt inspection and refueling operations that required the Fuel Handling Supervisor (FHS) to be trained and certified. During Unit 3 new fuel receipt inspections and refueling operations unqualified senior reactor operators (SRO) were allowed to supervise fuel handling activities. The unqualified SROs were subsequently re-qualified or not allowed to supervise fuel handling activities until qualified. This issue was entered into the licensee's corrective action program as problem evaluation reports 220410 and 220791.

This finding was determined to be of greater than minor significance because it was associated with the Barrier Integrity Cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the use of unqualified FHS(s) to supervise new fuel receipt inspection and core refueling operations would reduce the level of assurance that fuel handling activities were accomplished safely and error free to prevent inadvertent fuel damage. The finding was evaluated and determined to be of very low safety significance using Inspection Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, Attachment 1, Phase 1 Operational Checklists, Checklist 7, because it did not involve any human performance errors that resulted in fuel assembly damage, inappropriate core alteration, loss of reactor coolant and/or spent fuel pool inventory, or reduction of any safe shutdown mitigation capability. The cause of this finding was directly related to the cross-cutting aspect of Procedural Compliance in the Work Practices component of the Human Performance area because neither the night shift FHS or relief FHS(s) complied with the operating procedure requirements that all personnel supervising new fuel receipt inspections and/or refueling operations must be qualified [(H.4(b)]. (Section 1R20.1.1)

B. Licensee-Identified Findings

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

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full Rated Thermal Power (RTP) the entire report period except for two planned downpowers. On January 17, 2010, a planned downpower to approximately 95 percent RTP was conducted to perform control rod exercise surveillance and was returned to full RTP the same day. Then again on March 27, 2010, a planned downpower to approximately 70 percent RTP was conducted to adjust the control rod pattern. Unit 1 returned to full RTP the same day.

Unit 2 operated at essentially full RTP the entire report period except for one planned shutdown, one planned downpower, and an unplanned downpower. On January 10, 2010, a planned shutdown was conducted to repair reactor coolant seat leakage past the Loop I core spray injection check valve and inboard injection valve. The unit was restarted on January 15, 2010, and returned to full RTP on January 17, 2010. On February 21, 2010, a planned downpower was conducted to approximately 70 percent RTP to repair the 2B2 moisture separator high level dump valve, clean water boxes, and adjust the control rod pattern. The unit returned to full RTP on February 22, 2010. Then on March 8, 2010, an unplanned downpower to approximately 22 percent RTP was conducted in order to take the Unit 2 main generator turbine (MTG) off-line due to the discovery of a damaged bushing on the low-side of the 2B Unit Station Service Transformer (USST). After replacing all the low-side bushings on the 2B USST, the MTG was synchronized to the grid and Unit 2 returned to full RTP on March 14, 2010.

Unit 3 operated at essentially full RTP the entire report period except for two planned downpowers and a planned shutdown. On January 15, 2010, a planned downpower to 83 percent RTP was conducted to adjust the control rod pattern and clean water boxes. The unit was restored to full RTP on January 16, 2010. On February 3, 2010, a planned downpower was conducted to 95 percent RTP to remove extraction steam from 3A1 and 3A2 reactor feedwater heaters to facilitate repairs on the 3A1 moisture separator normal level control valve. The unit was returned to full RTP the same day. On February 27, 2010, the unit was shutdown for the Unit 3 Cycle 14 (U3R14) refueling outage (RFO) and remained in a shutdown condition for the remainder of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors conducted three partial equipment alignment walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, while the other train or system inoperable or out of service. The inspectors reviewed the functional systems descriptions, Updated Final Safety Analysis Report (UFSAR), system operating procedures, and Technical Specifications (TS) to determine correct system lineups for the current plant conditions. The inspectors performed walkdowns of the systems to verify that critical components were properly aligned and to identify any discrepancies which could affect operability of the redundant train or backup system. Documents reviewed are listed in the Attachment to this report.

- Unit 3 3EA Emergency Diesel Generator (EDG) System
- Unit 3 High Pressure Coolant Injection (HPCI) System
- Unit 3 Division I Residual Heat Removal (RHR) Aligned for Shutdown Cooling

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Walkdowns

a. Inspection Scope

The inspectors reviewed licensee procedures, Standard Programs and Processes (SPP)-10.10, Control of Transient Combustibles, and SPP-10.9, Control of Fire Protection Impairments, and conducted a walkdown of the seven fire areas (FA) and fire zones (FZ) listed below. Selected FAs/FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure SPP-10.9. Furthermore, the inspectors reviewed applicable portions of the Site Fire Hazards Analysis Volumes 1 and 2 and Pre-Fire Plan drawings to verify that the necessary fire fighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, were in place.

- Unit 1 Control Building Elev. 593 (FA-16)
- Unit 2 Control Building Elev. 593 (FA-16)
- Unit 3 Control Building Elev. 593 (FA-16)
- Units 1, 2 and 3 Control Building Elev. 617 (FA-16)
- Unit Common Control Building Elev. 606, Cable Spreading Rooms (FA-16)
- Unit 2 Reactor Building Elevs. 519, 541 and 565 west of column line R11 (FZ 2-1)
- Unit 3 Reactor Building, Elev. 593, and RHR Heat Exchanger rooms (FZ 3-3)

b. Findings

No findings of significance were identified.

1R06 Internal Flood Protection Measures

a. Inspection Scope

The inspectors performed a review of the flood protection measures for the residual heat removal service water (RHRSW) intake structure. The inspectors specifically examined plant design features and measures intended to protect the plant and its safety-related equipment from internal flooding events, such as flood level switches; floor drain sump level instrumentation; and bulkhead watertight doors, curbing, and wall penetrations. The inspectors performed walkdowns of risk-significant areas that included susceptible systems and equipment in the A, B, C, and D RHRSW Pump Rooms, to verify the condition of flood-mitigation features such as flood protection door seals, area level switches, room sumps and sump pumps, conduit seals, and instrument racks that might be subjected to flood conditions.

The inspectors reviewed applicable sections of licensing basis documents such as the UFSAR; General Design Criteria BFN-50-C-7105, Pipe Rupture, Internal Missiles, Internal Flooding and Vibration Qualification of Piping; TS requirements and Bases; and the Probabilistic Safety Assessment Internal Flooding Notebook. The inspectors also reviewed applicable emergency operating instructions (EOIs), and annunciator response procedures (ARPs) for mitigating and responding to flooding events to verify that licensee actions were consistent with the plant's licensing and design basis. Additionally, the inspectors also reviewed a sampling of the licensee's corrective action documents with respect to flood-related items to verify that problems were being identified and corrected. Furthermore, the inspectors reviewed selected completed preventive maintenance (PM) procedures, work orders (WO), and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Non-Destructive Examination Activities and Welding Activities

a. Inspection Scope

From March 8 to March 12, 2010, the inspector observed and 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 inspectors' activities consisted of an on-site review of nondestructive examination (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: 1995 Edition with 1996 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards. For Browns Ferry Unit 3, this was the first outage of the third period of the third interval. The inspectors also reviewed a sample of inspection activities associated with components that were outside the scope of ASME Section XI requirements which were performed in accordance with commitments to follow industry guidance documents, such as the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

The inspectors' 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) of the N-10-1 SLCS nozzle weld
- Magnetic particle testing of pipe weld 3-47B452-1557-IA

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

- Welding package for work order #10661559 - Welded replacement of valve BFN-3-RTV-067-0301A, RT VLV TO PI-67-81A

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Internal Inspections

a. Inspection Scope

The inspector reviewed activities related to the planned repair and modification of select jet pump restrainer assemblies. For some assemblies, larger-than-allowable gaps between the jet pumps and set screws were seen during the visual examinations. The inspectors verified that planned repairs were being completed in accordance with BWRVIP requirements.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems that were identified by the licensee and entered into the corrective action program (CAP). The inspectors reviewed these corrective action documents (i.e., problem evaluation reports (PER)) to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. This review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The PERS reviewed by the inspectors are listed in the following attachment.

b. Findings

No findings of significance were identified

1R11 Licensed Operator Requalification

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On January 19, 2010, the inspectors observed an as-found licensed operator requalification simulator evaluation for an operating crew per Unit 2 Simulator Evaluation Guide OPL177.078, SRV Stuck Open, Earthquake, Loss of LPCI MG, Feedwater Line Break, Unisolable Steam Line Break on RCIC, Loss of High Pressure Makeup, Inability to Maintain RPV Water Level Above TAF, and Emergency Depressurization (C1).

The inspectors specifically evaluated the following attributes related to the operating crews' performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of Abnormal Operating Instructions (AOIs), and Emergency Operating Instructions (EOIs)
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedures (EPIP)
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the Unit Supervisor and Shift Manager

The inspectors attended the post-examination critique to assess the effectiveness of the licensee evaluators, and to verify that licensee-identified issues were comparable to issues identified by the inspector. The inspectors also reviewed simulator physical fidelity (i.e., the degree of similarity between the simulator and the reference plant control room, such as physical location of panels, equipment, instruments, controls, labels, and related form and function).

b. Findings

No findings of significance were identified.

.2 Annual Review of Licensee Requalification Examination Results

a. Inspection Scope

On December 17, 2009, the licensee completed the comprehensive biennial requalification written examinations and annual requalification operating tests required to be administered to all licensed operators in accordance with 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

.1 Routine

a. Inspection Scope

The inspectors reviewed three specific equipment issues listed below for structures, systems and components (SSC) within the scope of the Maintenance Rule (MR) (10CFR50.65) with regard to some or all of the following attributes: (1) Appropriate work practices; (2) Identifying and addressing common cause failures; (3) Scoping in accordance with 10 CFR 50.65(b) of the MR; (4) Characterizing reliability issues for performance monitoring; (5) Charging unavailability for performance monitoring; (6) Balancing reliability and unavailability; (7) Trending key parameters for condition monitoring; (8) System classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); (9) Appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2); and (10) Appropriateness and adequacy of (a)(1) goals and corrective actions (i.e., Ten Point Plan). The inspectors also compared the licensee's performance against site procedure SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; and SPP 3.1, Corrective Action Program. Furthermore, the inspectors reviewed, as applicable, WO, surveillance tests, PERs, cause determination evaluations (CDE) system health reports, engineering evaluations, and/or Maintenance Rule Expert Panel (MREP) meeting minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met.

- 4KV Unit Board and Shutdown Board Safety Related Loads Exceeded Unreliability Performance Criteria
- Emergency Equipment Cooling Water (EECW) Pump Exceeded Unavailability Performance Criteria
- Unit 2 and 3 Reactor Core Isolation Cooling (RCIC) and HPCI Controllers Exceeded Unreliability Performance Criteria

b. Findings

One finding of significance was identified.

Introduction: The inspectors identified a Green noncited violation (NCV) of 10 CFR 50.65(a)(2) for failure to demonstrate the A3 EECW pump performance was being effectively controlled through the PM program, or to place the system in 10 CFR 50.65(a)(1) status due to increased pump unavailability beyond the established performance criteria.

Description:

In January of 2009, the upper shaft of the A3 EECW pump was replaced due to shaft wear at the packing gland. This resulted in 134 hours of unavailability. During this replacement, the upper shaft bushing was found to be worn and out of tolerance so it was also replaced. During the post maintenance testing (PMT) of the A3 EECW pump on January 25, 2009, the pump was returned to service but placed in alert status due to high vibrations.

On March 20, 2009, a knocking sound developed from the shaft of the A3 EECW pump, and on March 24, 2009, the pump was declared out of service. The pump was subsequently replaced in April 2009 and returned to service. Replacement of the A3 EECW pump resulted in 295 hours of unavailability. The pump failed the following IST flow surveillance, and an additional 59 hours of unavailability was accrued to trouble shoot the cause of the low flow. Cause Determinations and Evaluation 760 was then generated in April 2009 to address the cause of the unavailability during March and April 2009. This CDE concluded that all the pump unavailability was due to a manufacturer's defect. However, the CDE attributed the cause of the failure to replacing just the upper shaft bushing, without also replacing the lower shaft bushings, when the upper shaft was replaced in January 2009. This resulted in uneven tolerances in the shaft that led to the knocking sound and high vibrations.

In May 2009, the A3 EECW pump was again replaced due to the continued inability to achieve adequate flow during the IST flow surveillance. This replacement resulted in 514 hours of unavailability. No CDE was generated at that time to evaluate the unavailability.

In October 2009, the inspectors identified that the licensee had failed to recognize and evaluate the increased unavailability of the A3 EECW pump which had exceeded its unavailability performance criteria by more than double. In response to the inspectors' concern, the licensee initiated PER 204651 to identify instances where unavailability had increased above the threshold and to generate CDEs for the MREP to use to evaluate the need to place the pumps in 10 CFR 50.65(a)(1) status.

In response to PER 204651, CDE 850 was generated in January 2010 to address the cause of the unavailability in May 2009. This CDE determined the increased unavailability was due to planned maintenance on the EECW north header to repair a supply valve to the Loop II Core Spray Room Cooler. This maintenance only accounted for 43 hours of the total unavailability in May 2009.

Both CDE 760 and CDE 850 recommended that the pump remain in 10 CFR 50.65 (a)(2) status since the majority of the unavailability was supposedly due to the motor replacement in May 2008 and a strainer failure in July 2008, both of which were already, or had been, in 10 CFR 50.65(a)(1) status. These CDEs went on to state that the only unavailability on the pump that should be considered for 10 CFR 50.65(a)(1) status was the upper shaft replacement and this activity was not enough by itself to exceed the unavailability performance criteria. On February 11, 2010, the MREP reviewed the MR status of the A3 EECW pump and determined it should remain in 10 CFR 50.65(a)(2) status based on these CDEs. The inspectors questioned the accuracy of these CDEs. In response, the licensee re-examined past maintenance performed on the A3 EECW pump and subsequently revised CDE 850 to more accurately reflect the actual unavailability hours attributed to the pump for maintenance in May 2009. On March 30, 2010, the MREP met again to conduct an additional review of the A3 EECW pump unavailability, including the revised CDE. This time the MREP concluded that the EECW pumps should be placed in 10 CFR 50.65(a)(1) status.

Analysis: The licensee's failure to effectively monitor A3 EECW pump unavailability and adequately evaluate the causes of the unavailability to determine the impact on the 10 CFR 50.65(a)(2) performance criteria was considered a performance deficiency. This finding was determined to be of greater than minor significance because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone, and adversely affected the cornerstone objective of ensuring availability and reliability of systems designed to respond to initiating events to prevent undesirable consequences. More specifically, the licensee failed to demonstrate effective control of EECW system availability through appropriate PM. According to NRC Inspection Manual Chapter (IMC) 0609.04, Phase I - Initial Screening and Characterization of Findings, this finding was determined to be of very low safety significance (Green) because it did not lead to an actual loss of a system safety function or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

The cause of this finding was directly related to the cross cutting aspect of Thorough Evaluation of Identified Problems in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee did not adequately evaluate the causes of the A3 EECW pump unavailability and thereby failed to correctly determine the impact on the 10 CFR 50.65(a)(2) unavailability performance criteria [P.1(c)].

Enforcement: 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants, Paragraph (a)(2) states, in part, that the monitoring specified in paragraph (a)(1) is not required where it has been demonstrated the performance or condition of a system, structures and components is being effectively controlled through the performance of appropriate PM such that the system, structures and components remains capable of performing its intended function. Paragraph (a)(1) required, in part, that licensees shall monitor the performance or condition of system, structures and components within the scope of the rule against licensee-established goals in a manner sufficient to provide reasonable assurance the system, structures and components are capable of fulfilling their intended safety functions. Contrary to the above, the licensee failed to adequately evaluate the unavailability of the A3 EECW pump and consequently failed to demonstrate that the performance or condition of the A3 EECW pump had been effectively controlled through the conduct of appropriately scheduled PM, without the monitoring requirements specified in 10 CFR 50.65, Paragraph (a)(1) being implemented. However, because this finding was of very low safety significance (Green) and has been entered in the licensee's corrective action program as PER 223404, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. This NCV is identified as 05000259, 260, 296/2010002-01, Failure to Adequately Monitor Performance of the A3 EECW Pump as Required by 10 CFR 50.65.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For planned online and shutdown work, and/or emergent work, that affected the combinations of risk significant systems listed below, the inspectors reviewed three licensee maintenance risk assessments and actions taken to plan and control work activities to effectively manage and minimize risk. The inspectors verified that risk assessments and applicable risk management actions (RMA) were conducted as required by 10 CFR 50.65(a)(4) and applicable plant procedures such as SPP-7.1, Online Work Management; 0-TI-367, BFN Equipment to Plant Risk Matrix; SPP-7.3, Work Activity Risk Management Process; and SPP-7.2, Outage Management. The inspectors also evaluated the adequacy of the licensee's risk assessments and verified implementation of RMAs.

- 3A EDG, Unit 3 RCIC Pump and A2 RHRSW Pump Out of Service (OOS)
- 3B EDG, 3A Control Rod Drive (CRD) Pump, all four Control Air Compressors and three Raw Cooling Water Pumps OOS
- Unit 3 RHR Loops I and II, Core Spray (CS) Loop II, and 500 KV Offsite Power OOS During U3R14 RFO

b. Findings

No findings of significance were identified

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the seven operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedures SPP-3.1.3, Regulatory Screening, and NEDP-22, Functional Evaluations, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors examined the implementation of compensatory measures to verify that they achieved the intended purpose and that the measures were adequately controlled. The inspectors also reviewed PERs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- Units 1, 2 and 3 Emergency Core Cooling System (ECCS) Gas Accumulation (PERs 171845 and 174307)
- Unit 2 RHR Drywell Spray Line Void (PER 210961)
- Unit 2 RHR Loop II Injection Valve Reactor Coolant Seat Leakage (PER 210437)

- Unit 3 RCIC Flow Oscillations (PER 200183)
- 3A and 3D EDG Heat Exchanger EECW Low Flow (PERs 213088, 213374, and 213224)
- Unit 1 Automatic Depressurization System (ADS) Single Failure Impact on Loss of Coolant Accident (LOCA) Analyses (PERs 213059 and 213060)
- Unit 3 Auxiliary Decay Heat Removal (ADHR) Single Loop Operation (PERs 218258 and 222801)

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification listed below to verify regulatory requirements were met, along with procedure SPP-9.5, Temporary Alterations. The inspectors also reviewed the associated 10 CFR 50.59 screening and evaluation and compared each against the UFSAR and TS to verify that the modification did not affect operability or availability of the affected system. Furthermore, the inspectors walked down the modification to ensure that it was installed in accordance with the modification documents, and reviewed post-installation and removal testing to verify that the actual impact on permanent systems was adequately verified by the tests.

- TACF 3-10-002-067, Installation of Temporary Differential Pressure Gauges to Measure EECW Flow to Unit 3 Diesel Generators

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors witnessed the PMTs for the six maintenance activities listed below to verify that procedures and test activities confirmed SSC operability and functional capability following maintenance. The inspectors also reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors specifically reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). Furthermore, the inspectors verified that

PMT activities were conducted in accordance with applicable WO instructions, or procedural requirements, including SPP-6.3, Pre-/Post-Maintenance Testing, and MMDP-1, Maintenance Management System. The inspectors also reviewed any problems associated with PMTs to verify they were identified and entered into the CAP.

- 3B EDG Turbocharger Replacement per WO 08-712742-004
- Unit 3 CS Loop I Motor-Operated Valve (MOV) Breaker and Motor Inspections per WOs 09-717569-000 and 09-717557-000
- Unit 3 RCIC Steam Supply Inboard Isolation Valve, 3-FCV-71-2, Repairs per WO 110729511
- Unit 2 CKV-75-26 Check Valve Repair per WO 09-718598-000
- Unit 3 ADHR Primary Heat Exchanger Replacement, Secondary Pump Motor Replacement, 480 VAC Breaker Inspections and Replacements, and Backup Diesel Generator Installation per WOs 06-721232-000/1, 10559895, 110707511, 110701172, and 09-720453-000
- Unit 3 EDG Heat Exchanger EECW Supply System Flush per WO 10521967

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 3 Cycle 14 Refueling Outage

a. Inspection Scope

Beginning on February 27 to the end of the report period on March 31, 2010, the inspectors examined critical U3C14 RFO activities to verify that they were conducted in accordance with TS, applicable procedures, and the licensee's outage risk assessment and management plans. Activities occurring after March 31, 2010 will be documented in the next inspection report, including the Unit 3 restart and review of Fatigue Management. Some of the more significant inspection activities conducted by the inspectors were as follows:

Outage Risk Assessment

Prior to the U3R14 RFO that began on February 27, the inspectors attended outage risk assessment team meetings and reviewed the Outage Risk Assessment Report to verify that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing an outage plan that assured defense-in-depth of safety functions were maintained. The inspectors also reviewed the daily U3R14 Refueling Outage Reports, including the ORAM Safety Function Status, and regularly attended the twice a day outage status meetings (i.e., STORM meetings). These reviews were compared to the requirements in licensee procedure SPP-7.2, Outage Management, SPP-7.3, Work Activity Risk Management

Process, and TS. These reviews were also done to verify that for identified increased risk significant conditions, due to equipment availability and/or system configurations, contingency measures were identified and incorporated into the overall outage and contingency response plan. Furthermore, the inspectors frequently discussed risk conditions and designated protected equipment with Operations and outage management personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process

The inspectors witnessed the shutdown and cooldown of Unit 3 in accordance with licensee procedures OPDP-1, Conduct of Operations; 3-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; and 3-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 3-OI-74, Residual Heat Removal System (RHR); 3-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Auxiliary Decay Heat Removal System Failures; and conducted walkdowns of the main control room panels and applicable in-plant systems and components to verify correct system alignment. In addition, the inspectors reviewed controls implemented to ensure that outage work was not impacting the ability of operators to operate spent fuel pool cooling, RHR shutdown cooling, and/or ADHR system. Furthermore, the inspectors conducted several walkdowns of the ADHR system during operation with the fuel pool gates removed.

Critical Outage Activities

The inspectors examined outage activities to verify that they were conducted in accordance with TS, licensee procedures, and the licensee's outage risk control plan. Some of the more significant inspection activities accomplished by the inspectors were as follows:

- Reviewed and walked down selected safety-related equipment clearance orders (e.g., hanging of Equipment Clearance Order 3-TO-2010-0003, Clearance 3-281-0001, for 3A 250V Reactor Motor Operated Valve Board breaker testing; and removal of Equipment Clearance Order 3-TO-2010-0003, Clearance 3-211-0002, to return 3EA 4kV Shutdown Board to service following breaker testing).
- Verified reactor coolant system (RCS) inventory controls, especially during evolutions involving operations with the potential to drain the reactor vessel (OPDRV) controlled per 3-POI-200.5
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) and TS compliance during the various shutdown modes of operation, and mode transitions

- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations and overall integrity
- Examined foreign material exclusion controls particularly in proximity to and around the reactor cavity, equipment pit, and spent fuel pool
- Conducted tours of the main control room, reactor building, turbine building refueling floor and drywell

Reactor Vessel Disassembly and Refueling Activities

The inspectors witnessed selected activities associated with reactor vessel disassembly, and reactor cavity flood-up and drain down in accordance with 3-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, on numerous occasions, the inspectors witnessed fuel handling operations during the two Unit 3 reactor core fuel shuffles performed in accordance with TS and applicable operating procedures, such as 0-GOI-100-3A, Refueling Operations (In Vessel), 0-GOI-100-3B, Operations in the Spent Fuel Pool Only, and 0-GOI-100-3C, Fuel Movement Operations During Refueling. Furthermore, the inspectors verified specific fuel movements as delineated by the Fuel Assembly Transfer Sheets (FATF) and observed part of the Unit 3 core verification in accordance with 0-GOI-100-3C.

Torus and Drywell Closeout

On March 28, 2010, the inspectors reviewed the licensee’s final closure of the Unit 3 suppression chamber (i.e., torus) in accordance with 3-GOI-200-2, Primary Containment Initial Entry and Closeout, and performed an independent detailed closeout inspection of the Unit 3 torus.

The inspectors conducted two independent detailed closeout inspections of the Unit 3 drywell prior to and ending on March 31, 2010. The inspectors also reviewed and verified the licensee’s conduct of 3-GOI-200-2.

Pre- Restart Activities

The inspectors also specifically conducted the following:

- Attended multiple restart Plant Oversight Review Committee (PORC) meetings

- Inspected heatup and pressurization of Unit 3 reactor pressure vessel in accordance with 3-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor pressure Vessel and Associated Piping; and 3-SR-3.4.9.1(2), Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring during In-service Hydrostatic or Leak Testing

- Inspected portions of control rod scram time testing in accordance with 3-SR-3.1.4.1, Scram Insertion Time

- Reviewed and verified completion of selected items of 0-TI-270, Refueling Test Program, Attachment 2, Startup Review Checklist, and SPP-7.2.3, Plant Startup Review/Checklists

Corrective Action Program

The inspectors reviewed the PERs generated during U3R14 RFO, especially those designated as “Restart”. The inspectors also reviewed the “U3R14 Restart PER” and “U3R14 Restart 91-18 Issues” Lists. Furthermore, the inspectors attended PER Screening Committee and Corrective Action Review Board meetings to verify that initiation thresholds, priorities, mode holds, operability concerns and significance levels were adequately addressed. Resolution and implementation of selected corrective actions of selected PERs were also reviewed by the inspectors and discussed with responsible outage management.

b. Findings

Several findings of significance were identified, as described below.

(1) Introduction: A Green NCV of TS 5.4.1.a was identified by the inspectors for the licensee’s failure to ensure only fully qualified Fuel Handling Supervisors (FHS) were allowed to supervise Unit 3 new fuel receipt inspection and refueling operations.

Description: During the U3R14 RFO, the inspectors observed Unit 3 reactor core refueling operations. In particular, the inspectors observed fuel handling activities on the Refueling Floor between the Unit 3 reactor and spent fuel pool (SFP) in accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling. These activities were to be conducted under the direction and continuous supervision of an FHS with an active senior reactor operator (SRO) license. The first reactor core fuel shuffle was performed on March 3 and 4, 2010, in preparation for replacing selected control rod (CR) blades and control rod drive mechanisms (CRDM). The CRDM exchange and CR blade replacements were completed on about March 10. Then the second, and final, fuel shuffle was completed on March 17, 2010. Portions of the first and second Unit 3 fuel shuffle, and core alterations, were observed by the inspectors.

The inspectors found that, according to Section 3.1.A of 0-GOI-100-3C, “All personnel performing duties of Fuel Handling Supervisor shall meet the requirements of SPP-10.8.” Also, according to Section 3.1.E of 0-GOI-100-2, New Fuel Operations, “All personnel performing the duties of Fuel handling Supervisor shall meet the qualifications specified by SPP-10.8”. The requirements and qualifications for an FHS were established and defined by SPP - 10.8, Nuclear Fuel Management, Appendix E, Fuel Handling and Receipt Inspection Certification and Training Program Requirements. In addition to SPP-10.8, more detailed guidance was provided by Operations Training Guide (OTG) 45, Refueling Activity Qualifications. According to both, SPP-10.8, Appendix E, Section 3.0, and OTG 45, Section 3.1, an FHS must fulfill certain prescribed qualification standards, and then be re-evaluated and recertified at least every three years. Also according to Sections 3.A and D. of SPP-

10.8, a fully qualified FHS may be relieved for short periods of time by a “partially” qualified FHS during refueling operations, but not for new fuel receipt inspections.

On March 5, after observing the first core fuel shuffle, the inspectors requested information from the licensee to verify the qualifications of the full-time on-shift FHSs and the part-time relief FHSs. The licensee subsequently concluded that the qualifications for the night shift FHS had expired, and none of the short-time relief FHS(s) used during the first fuel shuffle were not qualified per SPP-10.8. The qualifications of the night shift FHS had expired not only for Unit 3 refueling operations, but also for the several weeks of Unit 3 new fuel receipt inspections that occurred prior to U3R14 RFO. Once this was recognized, the licensee promptly removed the night shift FHS from any further responsibilities for supervising fuel handling activities or core alterations, and then verified the existing qualifications of all other Operations personnel. No other Operations department qualification issues were identified. A review of the FHS qualification records by the inspectors confirmed that the qualifications for the night shift FHS had expired on November 29, 2009. In addition, even though the qualification requirements for short-time relief FHS(s) were not as restrictive as the full time FHS, the minimum requirements of SPP-10.8 had not been met for any of the relief FHS(s). The licensee also took actions to ensure any future relief FHS(s) would meet the requirements of SPP-10.8 before relieving a fully qualified onshift FHS. Furthermore, the licensee entered these issues into their CAP as PER 220410 for the expired qualifications of the night shift FHS, and PER 220791 for the lack of adequate qualifications for the relief FHS(s).

The night shift FHS was subsequently re-qualified, and re-certified, on March 11, 2010. Also, measures were taken to ensure all relief FHS(s) fulfilled the requirements of SPP-10.8 prior to relieving a fully qualified FHS during Unit 3 fuel handling activities.

<u>Analysis</u>: The conduct of Unit 3 new fuel receipt inspections and refueling activities under the supervision of unqualified FHS(s) was a performance deficiency. This finding was determined to be of greater than minor significance because it was associated with the Barrier Integrity Cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the use of unqualified FHS(s) to supervise new fuel receipt inspection and core refueling operations would reduce the level assurance that fuel handling activities were accomplished safely and error free. This finding was determined to be of very low safety significance (Green) using IMC 609, Appendix G, Shutdown Operations Significance Determination Process, Attachment 1, Phase 1 Operational Checklists, Checklist 7, BWR Refueling Operation with RCS Level > 23’, because it did not involve any human performance errors that resulted in fuel assembly damage, inappropriate core alteration, loss of reactor coolant and/or spent fuel pool inventory, or reduction of any safe shutdown mitigation capability.

The cause of this finding was directly related to the cross-cutting aspect of Procedural Compliance in the Work Practices component of the Human Performance area, because neither the night shift FHS and relief FHS complied with the procedure requirements of 0-GOI-100-3C and 0-GOI-100-2 that all personnel supervising new fuel receipt inspections and/or fuel handling operations must be qualified [H.4.(b)].

Enforcement: Technical Specification 5.4.1.a. required that written procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained. General plant operating procedures for Refueling and Core Alterations were specifically listed as recommended procedures by Sections 2.I of RG 1.33, Appendix A. General operating instructions 0-GOI-100-3C and 0-GOI-100-2 were established by the licensee for fuel inspection and handling activities prior to and during refueling operations. Section 3.1.A of 0-GOI-100-3C, and Section 3.1.E of 0-GOI-100-2, required all personnel performing duties of an FHS to meet the requirements of SPP-10.8. According to SPP-10.8, Appendix E, Section 3.0, and OTG 45, Section 3.1, an FHS must meet certain prescribed qualification standards, and then be re-evaluated and recertified at least every three years. Contrary to these requirements, during the months of January through March, 2010, the night shift FHS supervised Unit 3 new fuel receipt inspection and refueling activities after his qualifications had expired; and, on March 3, 2010, at least one relief FHS supervised ongoing refueling operations without being qualified. However, because the finding was of very low safety significance (Green) and has been entered into the licensee's CAP as PER 2220410 and 220791, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy. This NCV is identified as NCV 05000296/2010002-02, New Fuel Receipt Inspection and Refueling Operations Supervised By Non-qualified Senior Reactor Operators.

(2) Introduction: A Green NCV of TS 5.4.1.a was identified by the inspectors for the reactor operators failure to follow 3-SR-3.4.9.1(2), Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring during In-service Hydrostatic or Leak Testing, to ensure all required temperatures were being monitored and verified to meet TS 3.4.9, RCS Pressure and Temperature Limits.

Description: At the end of the U3R14 RFO, the licensee conducted a reactor pressure vessel (RPV) in-service leak test in accordance with 3-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping (ASME Section III, Class I and II). In order to meet TS surveillance requirement (SR) 3.4.9.1 during this test, the licensee monitored both RPV and RCS temperatures, and pressure, in accordance with 3-SR-3.4.9.1(2), Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring during In-service Hydrostatic or Leak Testing. During the RPV leak test, Attachment #2, RPV Temperature Monitoring, of 3-SR-3.4.9.1(2) was used by the operators to monitor, record, and verify all required RPV and RCS temperatures were at or to the right of the curves on TS Figure 3.4.9-2, Pressure/Temperature Limits for Reactor In-Service Leak and Hydrostatic Testing.

On March 26, 2010, the inspectors inspected the conduct of the Unit 3 RPV leak test while the operators were maintaining reactor pressure between 1044 -1064 psig per 3-SI-3.3.1.A. The inspectors also reviewed the 3-SR-3.4.9.1(2), Attachment #2, data sheets and procedure step signoffs from the start of the test, to 0700 March 26. Both the nightshift and dayshift operators had signed off in Attachment #2 of 3-SR-

3.4.9.1(2) every 30 minutes stating that all monitored temperatures were correct and to the right of Curve #1 and #2 of TS Figure 3.4.9-2. However, the inspector identified several significant discrepancies. Contrary to the operator's data entry and verification signatures at 0100, 0130, and 0200 CDT for Step 7.0[5.11], the temperatures recorded on Attachment #2 for Reactor Vessel Beltline Region were actually below and to the left of Curve #2 of TS Figure 3.4.9-2. These temperatures were about one to two degrees below the temperature required to meet Curve #2 of TS Figure 3.4.9-2, and thereby failed to meet the acceptance criteria of 3-SR-3.4.9.1(2). However, the night shift operator and Unit Supervisor (US) did not recognize this data was outside the allowed limits of TS Figure 3.4.9-2. Consequently, the operator and US failed to understand the TS 3.4.9 Limiting Condition of Operation (LCO) requirements were not met and thus did not take the required TS actions. Furthermore, the oncoming day shift control room operating crew also did not recognize the temperature data indicated TS 3.4.9 LCO had been exceeded. In fact, the dayshift crew continued to record the incorrect RPV metal temperature until it was identified by the inspector. The licensee initiated PER 222844 to address this issue.

In a subsequent investigation, the licensee determined that the nightshift operator had made an error in selecting the temperature element (TE) for monitoring RPV beltline temperature. The temperature specified by Attachment #2 to be recorded by the operator for the beltline region (with drywell blowers secured) was supposed to be 3-TE-56-23 (point 8) of Reactor Vessel Metal Temperature Recorder 3-TR-56-4. But instead, the operator inappropriately selected 3-TE-56-4, which was actually measuring the temperature of a nut on one of the RPV head flange closure studs. The actual RPV beltline temperatures during the RPV in-service leak test were verified by the licensee to be within the allowed bounds of TS Figure 3.4.9-2, Curve #2.

Analysis: The improper execution of surveillance procedure 3-SR-3.4.9.1(2) by control room operators during the Unit 3 RPV In-service Leak Test was a performance deficiency. This finding was determined to be of greater than minor significance because it was associated with the Initiating Events Cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. More specifically, the lack of reactor operator attention, and US oversight, during the RPV in-service leak test, resulted in operator errors that adversely affected the operators' ability to monitor and verify RPV metal temperatures were within TS Figure 3.4.9-2 limits to preclude a low temperature overpressure event. The finding was determined to be of very low safety significance (Green) according to the IMC 609.04, Phase 1 screening worksheet, because it did not actually exceed the TS limit or adversely affect any mitigating systems.

The cause of this finding was directly related to the cross-cutting aspect of Human Performance and Error Prevention in the Work Practices component of the Human Performance area, because human performance errors by the control room operators resulted in selecting the wrong RPV metal temperature to monitor and not recognizing this temperature exceeded TS limits [H.4.(a)].

Enforcement: Technical Specification 5.4.1.a. required that written procedures recommended in RG 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained. Item 8.b(2)(s) of RG 1.33, Appendix A, recommended surveillance test and inspection procedures required by TS for Nuclear Steam Supply System Pressurization and Leak Detection. Surveillance procedure 3-SR-3.4.9.1(2) was established to monitor RPV and RCS temperatures as required by TS SR 3.4.9.1 during an RPV hydrostatic/in-service leak test. Contrary to the above, on March 26, 2010, the control room operators failed to adequately implement 3-SR-3.4.9.1(2) when they did not monitor the correct temperature for RPV beltline temperature (with drywell blowers OOS), and improperly signed off that the surveillance procedure acceptance criteria of step 7.0[5.11] was satisfied for complying with TS Figure 3.4.9-2, Curve #2. However, because the finding was of very low safety significance (Green) and has been entered into the licensee's CAP as PER 222844, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy. This NCV is identified as NCV 05000296/2010002-03, Operators Failed to Correctly Monitor and Assess RPV Beltline Temperatures During RPV Hydrostatic/In-service Leak Test.

(3) Introduction: A Green NCV of TS 5.4.1.a was identified by the inspectors for the licensee's failure establish an adequate procedure to ensure all relevant RPV metal temperatures were being monitored pursuant with TS SR 3.4.9.1, RCS Pressure and Temperature Limits.

Description: In order to fulfill the surveillance testing requirements of TS SR 3.4.9.1, during RCS in-service leak and hydrostatic testing, the licensee established 3-SR-3.4.9.1(2),. The TS Bases 3.4.9, RCS Pressure and Temperature (P/T) Limits, identified the following four RPV regions to be monitored against the TS required P/T curve operating limits: Closure Flange, Core Beltline, Upper Vessel, and Lower Vessel. During the Unit 3 RPV inservice leak test, operators specifically used Attachment #2, RPV Temperature Monitoring, of 3-SR-3.4.9.1(2) to monitor, record, and verify all required RPV and RCS temperatures were at or to the right of the P/T operating limit curves of TS Figure 3.4.9-2.

During the conduct of the Unit 3 RPV inservice leak test, the inspectors reviewed 3-SR-3.4.9.1(2), including Attachment #2. Following this review, the inspectors identified a number of procedural discrepancies, and procedural inconsistencies with TS Bases 3.4.9. Of these procedural issues, the inspectors noticed multiple editorial errors on the Attachment #2 table for recording RPV and RCS temperatures that created human performance challenges for the operators. In addition, the Closure Flange region was not specified by 3-SR-3.4.9.1(2) to ensure the temperature of this RPV region was adequately monitored. Although, the RPV "Shell Adjacent to Flange" temperature elements were specified, other temperature elements for the closure head studs and top head flange identified by TS Bases 3.4.9 were not addressed. Furthermore, the temperature element 3-TE-56-23 specified by 3-SR-3.4.9.1(2), and Attachment #2, for the Core Beltline region (with drywell blowers OOS) was not within the beltline region defined by TS Bases 3.4.9. Per TS Bases 3.4.9, the Core Beltline region of the RPV was the vessel location adjacent to the active fuel. The 3-TE-56-23 thermocouple was actually at the RPV support skirt to bottom head junction, located well below the region of active fuel. In fact, TS Bases 3.4.9 identified the support skirt as part of the non-beltline region (i.e., Lower Vessel). The licensee initiated PERs 223539 and 224778 to address these issues.

In response to the procedure deficiencies identified by the inspectors, the licensee performed a historical review of all the applicable Unit 3 RPV temperatures recorded by the plant computer. Based on this review, the licensee was able to determine that all relevant RPV temperatures were within the limits of TS Figure 3.4.9-2, Curve 2 during the Unit 3 RPV in-service leak test.

Analysis: The failure to provide the operators with a surveillance procedure (i.e., 3-SR-3.4.9.1(2)) that would adequately monitor and verify all relevant RPV metal temperatures pursuant to TS SR 3.4.9.1, as described by TS Bases 3.4.9, was a performance deficiency. This finding was determined to be of greater than minor significance, because it was associated with the Initiating Events Cornerstone attribute of Procedure Quality, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. More specifically, the procedure used by operator's to monitor RCS and RPV temperatures, during the RPV in-service leak test, lacked sufficient details to ensure all relevant RPV temperatures would be monitored to meet TS SR 3.4.9.1 which could increase the likelihood of a low temperature overpressure event. The finding was determined to be of very low safety significance (Green) according to IMC 609, Phase 1 screening worksheet, because it did not actually exceed the TS limit or adversely affect any mitigating systems.

The cause of this finding was directly related to the cross-cutting aspect of Complete and Accurate Procedures in the Resources Component of the Human Performance area, because surveillance procedure 3-SR-3.4.9.1(2) lacked sufficient details and guidance to ensure all relevant RPV metal temperatures would be monitored pursuant to TS SR 3.4.9.1 [H.2.(c)].

Enforcement: Technical Specification 5.4.1.a. required that written procedures recommended in RG 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained. Item 8.b(2)(s) of RG 1.33, Appendix A, recommended surveillance test and inspection procedures required by TS for Nuclear Steam Supply System Pressurization and Leak Detection. Surveillance test procedure 3-SR-3.4.9.1(2) was established to monitor RPV and RCS temperatures as required by TS SR 3.4.9.1 during an RPV hydrostatic or in-service leak test. Contrary to TS 5.4.1.a, the licensee did not adequately establish in surveillance procedure 3-SR-3.4.9.1(2) the requirements necessary to ensure all relevant RPV metal temperatures would be monitored during an RPV hydrostatic or in-service leak test pursuant to TS SR 3.4.9.1. However, because the finding was of very low safety significance and has been entered into the licensee's CAP as PERs 223539 and 224778, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy. This NCV is identified as NCV 05000296/2010002-04, Inadequate Surveillance Procedure To Ensure All Relevant RPV Metal Temperatures Were Monitored During RPV Hydrostatic/In-service Leak Testing

.2 Unit 2 Forced Outage Due To Repair Loop I Core Spray Injection Valves

a. Inspection Scope

On January 10, 2010, Unit 2 conducted a planned Mode 4 shutdown to repair Loop I CS injection check valve 2-CKV-75-26 and inboard injection flow control valve 2-FCV-75-25 due to excessive reactor coolant seat leakage that was causing elevated CS discharge piping temperatures. The licensee also examined the reactor coolant seat leakage past the Loop II RHR injection check and inboard injection valves. Following repairs to the CS valves, Unit 2 was restarted on January 15, 2010, and reached full RTP on January 17, 2010. During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment and outage management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors during this report period were as follows:

- Unit shutdown and cooldown per 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; 2-AOI-100-1, Reactor Scram; and 2-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring
- Outage risk assessment and management per SPP-7.2 and SPP-7.3
- Control and management of forced outage and emergent work activities per SPP-7.2
- Control of Cold Shutdown (Mode 4) conditions, and monitoring of critical plant parameters
- Plant Oversight Review Committee post-trip review and restart meetings in accordance with SPP-10.5, Plant Operations Review Committee

Drywell Closeout

On January 15, 2010, the inspectors reviewed the licensee's final closure of the Unit 2 Drywell in accordance with 2-GOI-200-2, and performed an independent detailed closeout inspection of the Unit 2 Drywell.

Restart Activities

The inspectors reviewed and witnessed portions of the Unit 2 reactor startup and power ascension activities performed in accordance with per 2-GOI-100-1A, Unit Startup; 2-SR-3.4.9.1(1); 2-GOI-100-12, Power Maneuvering; and 0-TI-464, Reactivity Control Plan Development and Implementation.

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 2 forced outage to verify that initiation thresholds, priorities, mode holds, and significance levels were appropriate, and all restart PERs were dispositioned as required.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions and/or reviewed completed test data for the following eight surveillance tests of risk-significant and/or safety-related systems to verify that the tests met TS surveillance requirements, UFSAR commitments, and in-service testing and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the SSCs were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement.

In-Service Tests:

- 2-SI-4.4.A.1(Comp), Standby Liquid Control Comprehensive Pump Test
- 3-SR-3.5.1.6(CSII), Core Spray Flow Rate Loop II

Routine Surveillance Tests:

- 3-SR-3.8.1.9(3B OL), Diesel Generator 3B Emergency Load Acceptance Test With Unit 3 Operating
- 3-SR-3.5.1.1(HPCI), Maintenance of Filled HPCI Discharge Piping
- 1-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration
- 3-SI-3.2.4(DG C), EECW Check Valve Test on Diesel Generator C
- 1-SR-3.3.1.1.16 (APRM-3), Average Power Range Monitor (APRM) Functional Test APRM-3

Containment Isolation Valve Test:

- 3-SI-4.7.A.2.G-3/71A, Unit 3 Primary Containment Local Leak Rate Test RCIC Turbine Steam Supply: Penetration X-10

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On February 10, 2010, the inspectors observed an Emergency Preparedness drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures to identify any weaknesses and deficiencies in classification, notification activities, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room and Technical Support Center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure and other applicable Emergency Plan Implementing Procedures (EPIP). The inspectors also attended the licensee critique of the drill to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying weaknesses.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2RS1 Radiological Hazard Assessment and Exposure Control

a. Inspection Scope

Radiological Hazard Assessment: The inspectors reviewed a number of radiological surveys, including those performed for airborne areas, of locations throughout the facility including the Unit 3 drywell, Unit 1, Unit 2, and Unit 3 reactor buildings, the turbine building, and the independent spent fuel storage installation (ISFSI). The inspectors also walked down those same areas and select radioactive material storage locations with a survey instrument, evaluating material condition, postings, and radiological controls. The inspectors observed jobs in radiologically risk-significant areas including high radiation areas and areas with, or with the potential for, airborne activity. The inspectors determined that the surveys were adequate in thoroughness and frequency for the identified hazards.

Instructions to Workers: During plant walk downs, the inspectors observed labeling and radiological controls on containers of radioactive material. The inspectors also reviewed radiation work permits (RWP) used for accessing high radiation areas and airborne areas, verifying that appropriate work control instructions and electronic dosimeter (ED) setpoints had been provided and to assess the communication of radiological control requirements to workers. For selected tasks, the inspectors attended pre-job briefings that reviewed RWP details with the workers. The inspectors reviewed selected ED dose and dose rate alarms, to verify workers properly responded to the alarms and that the licensee’s review of the events was appropriate. Through observation of pre-job RWP briefings and health physics technician coverage of workers, the inspectors determined the licensee had established adequate means to notify workers of changing radiological conditions.

Contamination and Radioactive Material Control: The inspectors observed the release of potentially contaminated items from the radiologically controlled area (RCA) and from contaminated areas such as the drywell. The inspectors also reviewed the procedural requirements for, and equipment used to perform, the radiation surveys for release. During plant walk downs, the inspectors evaluated radioactive material storage areas and containers, including satellite RCAs and the low level radwaste facility, assessing material condition, posting/labeling, and control of materials/areas. In addition, the inspectors reviewed the sealed source inventory and verified labeling, storage conditions, and leak testing of selected sources.

The inspectors walked-down the ISFSI facility, observing the physical condition of the casks, radiological postings, and barriers. The inspectors performed independent gamma radiation surveys of the area and reviewed gamma radiation surveys of the ISFSI facility performed by licensee personnel. Inspectors compared the independent survey results to previous surveys and against procedural and TS limits. The inspectors evaluated implementation of radiological controls, including labeling and posting, and discussed controls with health physics staff. Environmental monitoring results for direct radiation from the ISFSI were reviewed and inspectors observed the placement and physical condition of thermoluminescent dosimeters around the facility.

Radiological Hazards Control and Work Coverage: The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress associated with the U3R14 RFO. Established radiological controls were evaluated for selected tasks including recirculation pump motor replacement, reactor vessel head set, scaffolding, and control rod drive (CRD) hydraulic control unit (HCU) maintenance. The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results.

During walk downs with a radiation survey meter, the inspectors independently verified ambient radiological conditions were consistent with licensee performed surveys, RWPs, and pre-job briefings; observed the adequacy of radiological controls; and observed controls for radioactive materials stored in the spent fuel pool. The inspectors also reviewed the procedural guidance for multi- and extremity badging. Select multi-badge packets were reviewed to verify consistency with procedural and regulatory guidance. For high radiation area tasks involving significant dose rate gradients, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure. The inspectors also reviewed and discussed selected whole-body count analyses conducted during 2009 and the U3R14 RFO. The inspectors reviewed RWPs for use in airborne areas, ensuring the prescribed controls were appropriate for the conditions as identified in radiological surveys and air samples. Electronic Dosimeter alarm set points and worker stay times were evaluated against area radiation survey results for drywell and refueling floor activities.

Risk-Significant High Radiation Area and Very High Radiation Area Controls: The inspectors discussed the controls and procedures for locked-high radiation areas (LHRAs) and very high radiation areas (VHRAs) with health physics supervisors and the radiation protection manager. The inspectors observed the issuance of LHRA keys and evaluated the storage, inventory, and handling of LHRA/VHRA keys.

During plant walk downs, the inspectors verified the posting/locking of LHRA/VHRA areas.

Radiation Worker Performance and Radiation Protection Technician Proficiency: The inspectors observed radiation worker performance through direct observation, via remote camera monitoring, and via telemetry. Jobs observed associated with the U3R14 RFO included recirculation pump motor replacement, reactor vessel head set, scaffolding, and CRD HCU maintenance. These jobs were performed in high radiation, airborne, and/or contaminated areas. The inspectors also observed health physics technicians providing field coverage of jobs and providing remote coverage.

Problem Identification & Resolution: Licensee CAP documents associated with radiation monitoring and exposure control were reviewed and assessed. This included a review of selected PERs related to radworker and health physics technician performance. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure SPP-3.1, Corrective Action Program, Rev. 18. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2RS01 of the Attachment.

Radiation protection activities were evaluated against the requirements of UFSAR Section 12; TS Sections 5.4 and 5.7; 10 Code of Federal Regulations (CFR) Parts 19 and 20; and approved licensee procedures. Radiological control activities for ISFSI areas were evaluated against 10 CFR Part 20, 10 CFR Part 72, and TS details. Records reviewed are listed in Section 2RS1 of the following Attachment.

The inspectors completed one sample, as described in Inspection Procedure (IP) 71124.01. The inspectors also completed the radiation protection line-item sample activities specified in IP 60855.1.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Barrier Integrity

RCS Activity and RCS Leakage

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the performance indicators (PIs) listed below, including procedure SPP-3.4, Performance Indicator Program. The inspectors also examined the licensee's PI data for the specific PIs listed below for the first through fourth quarters of 2009. The inspectors reviewed the licensee's data and graphical representations as reported to the NRC to verify that the data was correctly reported. The inspectors also validated this data against relevant licensee records (e.g., PERs, Daily Operator Logs, Plan of the Day, Licensee Event Reports, etc.), and assessed any reported problems regarding implementation of the PI program. Furthermore, the inspectors met with responsible plant personnel to review and discuss licensee records to verify that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors also used the Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to ensure that industry reporting guidelines were appropriately applied.

- Unit 1 RCS Activity
- Unit 1 RCS Leakage
- Unit 2 RCS Activity
- Unit 2 RCS Leakage
- Unit 3 RCS Activity
- Unit 3 RCS Leakage

b. Findings

No findings of significance were identified.

.2 Cornerstone: Occupational Radiation Safety

Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed PI data collected from April 1, 2009 through December 31, 2009, for the PIs listed below. The inspectors specifically reviewed CAP records to determine whether high radiation area, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred during the review period. In addition, the inspectors reviewed selected personnel contamination event data, internal dose assessment results, and ED alarms for cumulative doses and/or dose rates exceeding established set-points. The reviewed data were assessed against guidance contained in NEI 99-02. The documents reviewed relative to these PIs are listed in Sections 2RS1 and 4OA1 of the following Attachment.

- Unit 1 Occupational Exposure Control Effectiveness
- Unit 2 Occupational Exposure Control Effectiveness
- Unit 3 Occupational Exposure Control Effectiveness

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review of items Entered into the Corrective Action Program:

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 inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing daily PER report summaries, periodically attending Corrective Action Review Board (CARB) meetings and periodically attending PER Screening Committee (PSC) meetings.

.2 Focused Annual Sample Review of EECW System In-service Testing Code Requirements

a. Inspection Scope

The inspectors reviewed the specific corrective actions associated with PERs 175252, 175254, 175255, and 213180 which were previously identified by the inspectors regarding EECW system in-service testing (IST) that was required by the ASME Code for Operation and Maintenance (OM) of Nuclear Power Plants (Code of record: 1995, and 1996 Addenda).

The inspectors reviewed the licensee's corrective action plans and interviewed engineering personnel to assess the effectiveness and adequacy of the licensee's efforts to correct the inspector identified problems regarding licensee conformance with ASME OM Code requirements. The inspectors focused their review on the effectiveness of the licensee's corrective actions taken to address the conditions identified, including subsequent operability evaluations; the extent of condition analysis; and the prioritization of the corrective actions. Additionally, the inspectors evaluated these elements against regulatory requirements and the licensee's CAP.

This review constituted one problem identification and resolution (IP 71152) annual inspection sample.

b. Assessment and Observations

During the months of May and June 2009, the inspectors witnessed several tests on the A3 EECW pump. The pump had been replaced and returned to service in April 2009, but had subsequently failed its routine quarterly IST surveillance. The IST surveillance acceptance criteria for RHRSW and EECW pumps differed from other pumps in that the licensee used two sets of acceptance criteria for each pump based on river water temperature to account for the affect of seasonal variations of pump performance. These variations were due to changes in the impeller gap setting of the pump as the stainless steel pump shaft experienced thermal growth/contraction different from that of the carbon steel pump casing. The summer baseline was used when river temperature exceeded 65 degrees Fahrenheit (F), and a winter baseline was used when river water temperatures were below 65 F. During these tests, the inspectors raised several questions regarding whether the conduct of these tests were in compliance with the requirements of the ASME OM code. As a result of these questions, the licensee initiated several PERs as described below.

PER 175252

On June 29, 2009, PER 175252 was initiated to address a procedure problem with 0-SI-3.1.11, EECW Pump Baseline Data Acquisition and Evaluation, that was used for establishing two separate seasonally based IST baseline reference values (i.e., Summer and Winter) for EECW pump in-service testing. The reference points for these different baseline values were obtained from the routine quarterly surveillance, and/or comprehensive tests, and not from the original pre-service pump curve. Section ISTB 4.5, Establishment of Additional Set of Reference Values, of the ASME OM Code, required that any additional reference values must be determined from the original pre-service pump curve. The procedure used to develop the original pre-service pump curve was 0-TI-345, EECW Pump Curve Data Acquisition. However, 0-SI-3.1.11 only allowed the use of data from the normal quarterly surveillance, and/or biennial comprehensive flow tests, to establish the reference value(s), which did not establish a curve, but merely measured flow at a single point on the curve. The licensee's practice of not using the pre-service pump curve for establishing the EECW pump IST baseline reference values did not conform to the requirements of section ISTB 4.5 of the ASME OM Code. The corrective action plan for PER 175252 only identified one corrective action, which was to revise 0-SI-3.1.11 to allow the use of 0-TI-345 pump curve data in establishing the baseline values for future EECW system IST. However, it did not recognize that the existing and future establishment of additional reference values from other than the pre-service pump curve was in noncompliance with the ASME OM Code.

PER 175254

On June 29, 2009, PER 175254 was also initiated to determine if the river water temperature affects on EECW pump performance had the potential to mask pump degradation, and therefore not meet the code requirement for trending pump performance. Section ISTB 6.1 of the ASME OM code required that the parameters obtained from IST be trended. This PER had one corrective action, to perform an analysis to determine if river water temperature influences on pump performance were unduly masking pump degradation. This action was originally due on October 28, 2009. It was extended to December 4, 2009 and then extended again to June 30, 2010.

PER 175255

On June 29, 2009, another PER 175255 was initiated to evaluate whether a request for code relief was needed to address the problem that the permanently installed flow instrumentation used during EECW system IST did not meet the accuracy requirements of the code. Section ISTB 4.7.1 required that the instrumentation used to measure system flow rate shall be accurate to within ± 2 percent. In April 1998 the licensee identified that the flow rate instrumentation used for the EECW IST did not meet the accuracy requirements of the code (PER 35566) and had reduced the allowable acceptance criteria to less than code required to account for this inaccuracy. The inspector questioned whether this practice was allowed by the code or if a request for relief from the code accuracy requirement was needed. In response to inspector's concern (i.e., PER 175255), the licensee subsequently determined that the plant installed instrument accuracy was not in compliance with the ASME OM Code and a relief request would be required. This determination was made on September 23, 2009. However, the action to prepare a relief request was closed based on the decision to obtain more accurate flow measuring instrumentation that would meet the code requirements. The PER action, to evaluate the system configuration and install higher accuracy instrumentation, was subsequently closed on January 14, 2010, to PER action 156818-011. But the PER 156818-011 action was then closed on January 23, 2010, stating that the configuration and instrumentation had been reviewed and no changes should be made at this time. Although it was decided that a pump expert should be brought in to recommend solutions to this issue, no new corrective actions were added to the PER to ensure this pump expert was brought in. Furthermore, even though compensatory measures were in place that tightened the allowed flow, the corrective actions to obtain instrumentation that met the code, or request relief from the code, were closed without restoring full compliance. Neither this PER, or PERs 175252 and 175254, addressed the operability implications associated with the continued nonconforming conditions.

<u>PER 213180</u>

Based on further questions from the inspectors regarding the aforementioned EECW system code non-conformances, the licensee initiated PER 213180 on January 12, 2010 to assess the operability of the EECW system per NRC Inspection Manual Part 9900, Technical Guidance on Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety. This PER determined that both the instrumentation accuracy issues (PER 175255), and failure to use a pre-service pump curve to establish the required IST reference points (PER 175252), were issues of noncompliance with the ASME code. The licensee evaluated the impact of these non-conforming conditions on operability and determined that they did not adversely impact EECW system operability. However, the guidance in Part 9900 also required that these non-compliances be corrected at the first available opportunity, or provide an appropriate justification for a longer completion schedule. The licensee's initial corrective action plan for PER 213180 to restore full compliance with the ASME code had an assigned due date of February 11, 2011. Following further discussion with the inspectors, the licensee decided to establish an additional action to acquire higher accuracy gages, and revise all applicable surveillance procedures to include the more accurate flow measurement instrumentation, with a due date of June 17, 2010.

c. <u>Finding</u>

One finding of significance was identified.

Introduction: The inspectors identified a Green NCV of 10 CFR 50 Appendix B, Criterion XVI, for the licensee's failure to recognize and take appropriate corrective actions in a timely manner to restore compliance with the ASME OM code requirements for IST of the EECW system.

Description: In June 2009, the inspectors identified several issues involving noncompliance with ASME OM Code requirements during IST of EECW pumps (see Observations above). These issues involved the inappropriate use of instrumentation that did not meet the accuracy requirements of the code (PER 175255), and establishing seasonal baseline reference values based on EECW pump surveillance testing (quarterly or comprehensive test), rather than from the pre-service pump curve as required by the ASME code (PER 175252).

From June 2009 until January 2010, the licensee failed to adequately address the nonconforming nature of the IST issues, identified by the inspectors, as referenced in the guidance of NRC Regulatory Issue Summary 2005-20, Rev. 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." In January 2010, in response to inspector concerns, the licensee initiated PER 213180 to conduct a functional evaluation of these issues to determine their impact on EECW operability. This PER reconfirmed that the instrument accuracy issues described by PER 175255 was a nonconformance and established that the method of establishing the reference values for the EECW pumps described in 175252 was also a noncompliance with the ASME OM Code. However, PER 213180 did subsequently establish that these nonconforming conditions had not impacted the operability of the EECW pumps. The corrective action plan of PER 213180 was to revise the EECW system surveillance procedures to require the use of the more accurate instrumentation, and to establish single baseline reference values for each of the EECW and RHRSW pumps by February 11, 2011. However, the inspectors found that the corrective actions of PER 213180 failed to establish a corrective action plan that would correct the ASME OM Code nonconforming conditions at the first available opportunity, or provide an appropriate justification for a longer completion schedule. Following further discussion with the inspectors, the licensee did add a specific action to acquire higher accuracy flow measurement gages and revise all applicable IST surveillance procedures by June 17, 2010.

The licensee maintained a list of degraded or non-conforming conditions that would be reviewed by management on a regular basis and prior to each unit restart. In March 2010, the inspectors identified that none of the aforementioned PERs were included on the list of degraded or non-conforming conditions used as part of the licensee's Unit 3 restart readiness review conducted by PORC. The ASME OM Code non-conformances were subsequently added to the licensee's RIS 2005-20 list and were evaluated prior to the Unit 3 restart. However, during the U3R14 RFO restart PORC the licensee recognized there was insufficient time to resolve these issues prior to restart.

Analysis: The licensee's failure to conduct EECW system IST in conformance with ASME OM Code requirements was considered a performance deficiency. According to IMC 0612, Appendix B, this finding was considered more than minor because if left uncorrected it could become a more significant safety concern. In-service testing of the EECW system in conformance with the ASME OM Code provides assurance that degraded pump performance would be promptly detected and corrected. Failing to recognize and resolve these and other IST program deficiencies could lead to untimely detection of EECW pump degradation. According to IMC 0609.04, Phase I - Initial Screening and Characterization of Findings, this finding was determined to be of very low safety significance (Green) because it did not lead to an actual loss of system safety function or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

The cause of this finding was directly related to the cross-cutting aspect of Appropriate and Timely Corrective Actions in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee failed to take appropriate corrective actions to restore full compliance with the ASME OM Code requirements in a timely manner [P.1(d)].

Enforcement: 10 CFR 50, Appendix B, Criteria XVI, Corrective Action, requires in part, that measures be established to assure that conditions adverse to quality, such as non-conformances, are promptly identified and corrected. Contrary to the above, the licensee failed to promptly recognize, and then correct in a timely manner, non-conforming conditions involving the ASME OM code requirements for in-service testing of the EECW pumps identified on June 29, 2009. However, because this finding was of very low safety significance (Green) and has been entered into the licensee's CAP as PER 225844, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. This NCV is identified as 05000259, 260, 296/2010002-05, Untimely Corrective Actions to Restore Compliance of EECW Pump In-Service Testing with ASME OM Code Requirements.

.3 Focused Annual Sample Review of Snubber Testing Program and Multiple Test Failures During U3R14 RFO

a. Inspection Scope

The inspectors reviewed the licensee's Snubber Functional Test Scope for all specified snubber subgroups to be tested during U3R14 RFO to verify conformance with Technical Surveillance Requirements (TSR) 3.7.4.1 and 3.7.4.2 of the Unit 3 Technical Requirements Manual (TRM). The inspectors also reviewed the licensee's data base of Unit 3 snubbers scheduled to be tested during the outage. During initial implementation of the U3R14 RFO snubber test program, the licensee identified a functional failure of snubber BFN-3-SNUB-001-5041 in the 3A subgroup. The licensee initiated PER 220811, and developed a Snubber Functional Test Scope Expansion plan as required by TSR 3.7.4.3. During implementation of the expanded snubber test plan, the licensee identified a second functional failure in the 3A subgroup (i.e., BFN-3-SNUB-001-5033) and initiated PER 220943. Once again, the Snubber Functional Test Scope was expanded by 10 percent in accordance with TSR

3.7.4.3. During the implementation of the second snubber test plan expansion, the licensee identified another functional failure in the 3A subgroup (i.e., BFN-3-SNUB-001-5017) and initiated PER 221949. Pursuant to TSR 3.7.4.3, the licensee developed a third Snubber Functional Test Scope Expansion plan. The inspectors reviewed all three Snubber Functional Test Scope Expansion plans and verified their conformance with TSR 3.7.4.3 for increased sampling. Furthermore, the inspectors met with responsible component engineering personnel to discuss the overall snubber testing program, sample expansions, and each of the individual snubber functional failures.

The inspectors reviewed the engineering failure analyses for the three failed snubbers performed in accordance with 0-SI-4.6.H-2A, Functional Testing of Mechanical Snubbers, Attachment 3, Engineering Failure Analysis for Inoperable Snubbers. The inspectors also reviewed the operability impact on the associated Main Steam lines for the three failed snubbers performed in accordance with 0-SI-4.6.H-2A, Attachment 4, Supported System/Component Analysis for Inoperable Snubber. Furthermore, the inspectors verified licensee compliance with the requirements of TSR 3.7.4.3 and TSR 3.7.4.4 for performing engineering evaluations of failed snubbers and system/component operability, and incorporating the failure analysis results into their sample expansions.

The inspectors witnessed the functional testing of two balance of plant snubbers (i.e., 3-SNUB-008-2333 and 2300) in accordance 0-SI-4.6.H-2A. The inspectors also reviewed the successful test results of these snubbers. Furthermore, the inspectors interviewed the snubber testing contractors regarding the snubber test program and operation of snubber testing apparatus.

This review constituted one problem identification and resolution (IP 71152) annual inspection sample.

b. Observations and Findings

No Observations or findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) LER 05000296/2009-002, Inoperable High Pressure Coolant Injection System Due to Excessive Water in the Steam Line Drain

a. Inspection Scope

The inspectors reviewed Licensee Event Report (LER) 50-296/2009-002, dated January 11, 2010, and the applicable PER 207915, including associated apparent cause determination and corrective action plans.

On November 12, 2009, while securing the Unit 3 HPCI turbine following performance of 3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure, a high water level in the HPCI turbine exhaust steam drain pot alarm was received. Main control room operators responded in accordance with the applicable ARP which included dispatching assistant unit operators (AUO) to manually drain the condensate from the drain pot through the drain pot level switch instrument test drain. In excess of 80 gallons of condensate was drained from the HPCI turbine exhaust drain pot before the alarm cleared. A chemistry sample determined that the water was from the suppression pool. Operations declared the HPCI system inoperable and entered the applicable TS LCO actions. The licensee's apparent cause for the HPCI system inoperability was siphoning of water from the suppression pool via the HPCI drain pot drain line despite in-series check valves. Immediate corrective actions included closing, and administratively controlling, an isolation valve in the HPCI turbine exhaust drain line between the turbine exhaust drain pot and the suppression pool. Long term licensee actions taken or planned include permanently removing the HPCI turbine exhaust drain line from service on all three units to prevent siphoning.

b. Findings

No significant findings or violations of NRC requirements were identified. This LER is considered closed.

.2 (Closed) LER 05000260/2009-009, Inadvertent Isolation of the High Pressure Coolant Injection System During Testing Activities

a. Inspection Scope

The inspectors reviewed LER 05000260/2009-009 dated January 19, 2010, and the applicable PER 208627, including associated cause determination and corrective action plans.

On November 17, 2009 during performance of 2-SR-3.3.6.1.3 (3DFT), HPCI Steam Line Space High Temperature Functional Test, a Group 4 Primary Containment Isolation Signal was received which isolated steam to the HPCI turbine. During the testing a voltage was individually applied to each of the 16 temperature detector heating coils to ensure they function as designed. This voltage was applied through a test panel that connected to a common multi-pin connector. Subsequent examination of the plant computer data identified that two temperature elements had unexpectedly reached their high temperature setpoint, even though only one of the detectors was actually being tested, which caused the isolation logic to be met and thereby caused the HPCI steam line isolation. The surveillance test was promptly stopped and the test equipment disconnected. The alarm condition cleared and the isolation signal was reset to restore the HPCI system to operable status. The cause of this event was a faulty test connector due to a lack of insulation on the inner connector body. Also, the condition of the insulation on several other conductors within the assembly was insufficient to prevent inadvertent application of the test voltage to detectors that were not under test. The test connector was replaced and the surveillance was completed satisfactorily. Long term corrective actions taken or planned include inspection of the test connectors on Units 1 and 3, and replacement or repair of those connectors if conditions warrant.

b. Findings

No significant findings or violations of NRC requirements were identified. This LER is considered closed.

.3 (Closed) LER 05000260/2009-002-01, Leak In An ASME Code Class 1 Reactor Pressure Boundary Pipe

a. Inspection Scope

The original LER 50-260/2009-002 dated July 30, 2009, and applicable PER 172551, including the associated apparent cause analysis, and corrective action plans, were reviewed by the inspectors and documented in Section 4OA3.3 of NRC inspection report (IR) 05000260/2009004. As a result of this review, no significant findings or violations of NRC requirements were identified. However, the inspectors' review of the original LER identified several minor editorial errors that were discussed with the licensee. To address these errors the licensee initiated PER 201410. The inspectors reviewed this revised LER, dated January 4, 2010, and verified the apparent cause and associated PER 172551 corrective actions were not changed by the licensee as a result of the revisions to the LER.

b. Findings

No significant findings or violations of NRC requirements were identified. This revised LER is considered closed.

.4 (Closed) LER 05000260/2009-004-01, Technical Specification Shutdown Due to Rise in Unidentified Drywell Leakage

a. Inspection Scope

The original LER 50-260/2009-004 dated August 10, 2009, and applicable PERs 174596 and 173480, were reviewed by the inspectors and documented in Section 4OA3.6 of IR 05000260/2009004. However, during the review of the original LER, the inspectors identified numerous minor editorial deficiencies and errors for which the licensee initiated PER 205308. As part of the PER 205308 corrective actions, the licensee issued a revised LER 05000260/2009-004-01 on February 12, 2010. This LER was revised to correct and update the LER event description; expand the event cause, event analysis, safety assessment and corrective action sections; and update the Abstract. The revised LER also incorporated appropriate event description, root cause, and corrective actions for an unexpected subcritical reactor protection system (RPS) actuation that occurred shortly after the initial event. The inspectors reviewed the revised LER, and verified the root causes and previously identified corrective actions for the stuck open main steam relief valve (MSRV) were not substantially different, except for additional information provided regarding the licensee's extent of condition evaluation and related corrective actions. In addition, the inspectors determined the cause of the unexpected RPS actuation was not a performance deficiency. The RPS actuation occurred due to a loose RPS Channel B scram relay connection concurrently with a spurious spiking of an RPS Channel A intermediate range monitor.

b. Findings

No significant findings or violations of NRC requirements were identified. This revised LER is considered closed.

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 that 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 significant findings were identified.

.2 Follow-up On Alternative Dispute Resolution Confirmatory Orders (IP 92702)

a. Inspection Scope

During the inspection period the inspectors performed a follow-up review of TVA's completion of Confirmatory Order for Office of Investigation Report Nos. 2-2006-025 & 2-2009-003, item number 2;

"By no later than seven (7) calendar days after the issuance of this Confirmatory Order, a member of TVA's executive management responsible for the licensee's nuclear power plant fleet will, in writing, communicate TVA's policy, and the expectations of management, regarding the employees' rights to raise concerns without fear of retaliation in the context of this Confirmatory Order."

b. Findings and Observations

No significant findings or issues were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On April 1, 2010, the senior resident inspector presented the inspection results to Mr. James Randich and other members of the staff, who acknowledged the findings. A re-exit meeting was also conducted with Mr. Randich on April 9, 2010. During the course of the inspection, the licensee did provide the inspectors with certain documents for review that were considered to be of a proprietary nature. However, no proprietary information was included in this report.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) or Severity Level IV were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy, for being dispositioned as an NCV:

- Technical Specification 5.7.1.a, High Radiation Area, requires each entryway to a high radiation area (HRA) to be barricaded and posted as an HRA. Contrary to this, on August 28, 2009, the swing gate barricading an HRA in the Unit 2 SE Quad (541' elevation) was left propped open after completion of scaffold removal from the area by carpenters. The swing gate was found propped open by a health physics technician on the following shift. This was identified in the licensee's CAP as PER 200501. This finding was of very low safety significance because it did not involve As Low As Reasonably Achievable (ALARA) planning and controls, did not involve an overexposure, did not pose a substantial potential for overexposure, and the ability to assess dose was not compromised.

- Technical Specification 5.7.1.b, High Radiation Area, requires access to, and activities in, each HRA to be controlled by means of a RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s). Contrary to this, on February 1, 2010, two fix-it-now leak crew workers entered an HRA by ducking under a locked swing gate that was barricading entry into the Unit 2 SE Quad (519' elevation) from the torus. The workers were not on an RWP that allowed access into HRAs, were not wearing telemetry (as would have been required by the RWP), and had not been briefed on the radiological conditions in the area. The individuals were seen by nuclear assurance workers, who questioned them on being in an HRA without telemetry; the nuclear assurance workers then watched the leak crew workers duck back under the swing gate to exit. This was identified in the licensee's CAP as PER 215769. This finding was of very low safety significance because it did not involve ALARA planning and controls, did not involve an overexposure, did not pose a substantial potential for overexposure, and the ability to assess dose was not compromised.

- Technical Specification 5.7.1.b, High Radiation Area, requires access to, and activities in, each HRA to be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s). Contrary to this, on May 25, 2009, a Modifications boilermaker was handling bags of mop heads in a contaminated area during cavity decontamination while logged-on to a dose-control, non-contaminated area RWP. The mop heads were reading 1500 mrem/hour on contact and 500 mrem/hour at 30 cm; the rad-trash bag constituted a “mobile” high radiation area being controlled by a health physics technician. The boilermaker failed to check-in with the control point on the refuel floor, where he would have been briefed on the radiological conditions, radiation protection measures, and placed on the appropriate RWP for the task. When the boilermaker received a dose rate alarm, the health physics technician covering the job immediately removed him from the area, having recognized that the boilermaker could not be on the correct RWP since he had received an alarm. This was identified in the licensee’s CAP as PER 172081. This finding was of very low safety significance because it did not involve ALARA planning and controls, did not involve an overexposure, did not pose a substantial potential for overexposure, and the ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Berry, Component Engineering Manager
J. Black, Chemistry Manager
O. Brooks, Operations LOR Supervisor
S. Bono, Director of Engineering
M. Button, Maintenance Manager
J. Colvin, Engineering Programs Manager
R. Conner, Work Control Manager
M. Durr, Design Engineering Manager
J. Emens, Site Licensing Supervisor
A. Feltman, Emergency Preparedness Manager
F. Godwin, Licensing Manager
J. Keck, Reactor Engineering Manager
R. King, System Engineering Manager
D. Malinowski, Operations Training Manager
M. McAndrew, Operations Superintendent
J. McCarthy, Director Safety and Licensing
O. Miller, Operations Manager
J. Mitchell, Site Security Manager
J. Morris, Director Training
F. Nilsen, Site Engineer ISI/NDE
E. Quinn, Performance Improvement Manager
K. Polson, Site Vice President
J. Randich, Plant General Manager
R. Rogers, Director Project Management
P. Sawyer, Radiation Protection Manager
V. Schiavone, BWRVIP Coordinator
H. Smith, Fire Protection Supervisor
J. Underwood, Site Nuclear Assurance Manager
J. Walton, Health Physics Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Opened and Closed

05000259, 260, 296/2010002-01	NCV	Failure to Effectively Maintain Performance of the A3 EECW Pump as Required by 10 CFR 50.65(a)(2) (Section 1R12)

05000296/2010002-02	NCV	New Fuel Receipt Inspection and Refueling Operations Supervised By Non-qualified Senior Reactor Operators (Section 1R20.1.1)
05000296/2010002-03	NCV	Operators Failed to Correctly Monitor and Assess RPV Beltline Temperatures During RPV Hydrostatic/In-service Leak Test (Section 1R20.1.2)
05000296/2010002-04	NCV	Inadequate Surveillance Procedure To Ensure All Relevant RPV Metal Temperatures Were Monitored During RPV Hydrostatic/In-service Leak Testing (Section 1R20.1.3)
05000259, 260, 296/2010002-05	NCV	Untimely Corrective Actions to Restore Compliance of EECW Pump In-Service Testing with ASME OM Code Requirements (Section 4OA2.2)

Closed

05000296/2009-002-00	LER	Inoperable High Pressure Coolant Injection System Due to Excessive Water in the Steam Line Drain (Section 4OA3.1)
05000260/2009-009-00	LER	Inadvertent Isolation of the High Pressure Coolant Injection System During Testing Activities (Section 4OA3.2)
05000260/2009-002-01	LER	Leak in an ASME Code Class I Reactor Pressure Boundary Pipe (Section 4OA3.3)
05000260/2009-004-01	LER	Technical Specification Shutdown Due to Rise in Unidentified Drywell Leakage (Section 4OA3.4)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

3-OI-82, Standby Diesel Generator System, Rev. 90
3-OI-82, Attachment 1A, Standby Diesel Generator 3A Valve Lineup Checklist, Effective Date 03/05/07
3-OI-82, Attachment 2A, Standby Diesel Generator 3A Panel Lineup Checklist, Effective Date 03/05/07
3-OI-82, Attachment 3, Standby Diesel Generator Common Electrical Lineup Checklist, Effective Date 03/05/07
3-OI-82, Attachment 3A, Standby Diesel Generator 3A Electrical Lineup Checklist, Effective Date 03/05/07
3-OI-82, Attachment 4A, Standby Diesel Generator 3A Instrument Inspection Checklist, Effective Date 08/01/08
3-OI-73, High Pressure Coolant Injection System, Rev 43 and Attachments 1, 2 and 3
3-OI-74, Residual Heat Removal (RHR) System, Rev 95 and Attachments 1, 2 and 3
TS 3.4.8, Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown
TS 3.9.7, Residual Heat Removal (RHR) Shutdown Cooling System - High Water Level

Section 1R05: Fire Protection

Fire Protection Impairment Permit (FPIP) 10-2334, BFN-0-MTR-26-3 C Electric Fire Pump Motor
Fire Protection Impairment Permit (FPIP) 09-1920, BFN-0-APPR-SSD-LCO Use for Appendix R SSD LCOs
Fire Protection Impairment Permit (FPIP) 06-0175, Process Computer Room Halon Tanks and Control Panel OOS
Fire Protection Report, Volume 1, Section 2, Fire Hazards Analysis, Fire Area 16, Rev. 5
Fire Protection Report, Volume 2, Section IV.10, Pre-Plan Nos. CB1-593 and CB1-617, Rev. 5 and Rev. 4 respectively
Fire Protection Report, Volume 2, Section IV.11, Pre-Plan Nos. CB2-593 and CB2-617, Rev. 5
Fire Protection Report, Volume 2, Section IV.12, Pre-Plan Nos. CB3-593 and CB3-617, Rev. 5 and Rev. 4 respectively
Browns Ferry Nuclear Plant NEIL Comprehensive Report Dated October 12-14, 2004
SR 126905, 1.5” Conduit Missing Cable Pull Cover in Communications Room
SR 126889, Process Computer Room 1” Conduit Missing Halon Gas Seal
SR 127014, Correct Pre-fire Plans for CB1-593 and CB3-593
SR 127980, Penetration C25933884 is Not Labeled
SR 127999, Penetration Seals for Four 1” Conduits Not Labeled
SR 128896, Cable Configuration Needs to be assessed
SR 129028, Penetration Seal Labels Need Repainted
Fire Protection Report, Volume 2, Section IV.10, Pre-Plan No. CB1-606, Rev. 7
Fire Protection Report, Volume 2, Section IV.11, Pre-Plan No. CB2-606, Rev. 8
Fire Protection Report, Volume 2, Section IV.12, Pre-Plan No. CB3-606, Rev. 7
Fire Protection Report, Volume 2, Section IV.5, Pre-plan No. RX2-565, Rev. 9

Fire Protection Report, Volume 2, Section IV.4, Pre-plan No. RX2-519NW, Rev. 7
Fire Protection Report, Volume 2, Section IV.5, Pre-plan No. RX2-519SW, Rev. 9
Fire Protection Report, Volume 1, Section 2, Fire Hazards Analysis, Fire Zone 3-3, Rev. 6
Fire Protection Report, Volume 2, Section IV.9, Pre-Plan No. RX3-593, Rev. 8

Section 1R06: Internal Flood Protection Measures

0-AOI-100-3, Flood Above Elevation 558, Rev. 33
0-AOI-100-4, Breech of Wheeler Dam, Rev. 15
0-SIMI-23B, Scaling and Setpoint Document - RHRSW Pump Compt A/B Level High, Rev. 18
0-TI-171, RHRSW Sump Pump Flow Rate Test [completed 7/22/09], Rev. 6
1-ARP-9-22A, Panel 9-22, 1-XA-55-22A, Rev. 5
2-EOI-3, Secondary Containment Control. Rev. 11
Browns Ferry Nuclear Plant Unit 1 Probabilistic Safety Assessment Internal Flooding Notebook, Rev. 1
Calculation MD-Q0023-870149, RHRSW Pump Compartment Sump and Sump Pump Capacity, Rev. 14
Calculation MD-Q0023-890078, Pump Performance Analysis for New RHRSW Compartment Sump Pumps, Rev. 2
CCI-0-LS-23-087, RHR Service Water Pump Compartment Level Switches, Rev. 4
Drawing 0-37W205-5, Mechanical Pumping Station and Water Treatment – Piping and Equipment, Rev. 7
Drawing 2-45E779-18, Wiring Diagram 480v Shutdown Auxiliary Power, Rev. 27
EII-0-023-SSD001, Scaling and Setpoint Document – RHRSW Compartment Unwatering Pumps, Rev.6
EPI-0-000-SWZ006, Calibration and Inspection of Station Drainage and Intake Sump Pump Level Switches, Rev. 20
FSAR Section 1.2, Definition-Probable Maximum Flood, Amendment 21
FSAR Section 1.6, Plant Description-Flooding, Amendment 23
FSAR Section 2.4.2.2.3, Floods, Amendment 19
FSAR Appendix 2.4A, Browns Ferry Nuclear Plant Maximum Possible Flood, Amendment 22
FSAR Section 10.9, RHR Service Water System, Amendment 22
FSAR Section 12.2, Residual Principal Structures and Foundations, Amendment 22
General Design Criteria BFN-50-C-7105, Pipe Rupture, Internal Missiles, Internal Flooding and Vibration Qualification of Piping, Rev. 9
MPI-0-260-DRS001, Inspection and Maintenance of Doors [completed 10/16/09], Rev. 37
PER 205156, FSAR Basis for Floods Inadequate
SR 151709, Drain Hole in Base of C3 RHRSW /EECW Pump Clogged
SR 151730, Drain Hole in Base of D3 RHRSW /EECW Pump Clogged

Section 1R08: Inservice Inspection

Corrective Action Documents
PER 166464, A3 EECW Knocking Noise dated 3/24/09
PER 168770, Vendor documentation on RHR heat exchanger expansion joints.
PER 168907, EHC piping brace cracked dated 04/20/2009

PER 173662, FOUND BEARING HOUSING CRACKED, AND VALVE WOULD NOT CYCLE MANUALLY dated 06/19/2009

Procedures

N-VT-1, Rev. 44, Visual Examination Procedure for ASME Section XI Preservice and Inservice, 04/21/2009
N-PT-9, Rev. 33, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, 02/18/2009
N-UT-78, Rev. 5, PDI Generic Procedure for the Manual Ultrasonic Examination of Reactor Pressure Vessel Welds PDI-UT-6, 08/11/2008
N-UT-79, Rev. 2, PDI Generic Procedure for the Manual Ultrasonic Through Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds PDI-UT-7, 08/18/2008
N-UT-84, Rev. 0, Procedure for the Phased Array Ultrasonic Examination of Austenitic and Ferritic Pipe Welds, 10/21/2008
54-ISI-363, Rev. 5, Remote Underwater In-Vessel Visual Inspection of Reactor Pressure Vessel Internals, Components, and Associated Repairs in Boiling Water Reactors, 10/21/2008

Other Documents

CRP-ENG-SS-08-005, Snapshot Self-Assessment Report, 04/30/08
Corporate Engineering Welding Assessment Report, 08/03/2004
Browns Ferry Nuclear Plant (BFN) – Unit 2 – American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection, System Pressure Test, Containment Inspection (IWE), and Repair and Replacement Programs – Summary Reports (NIS-1 and NIS-2) for Cycle 14 Operation, 07/16/2007
ISI Report# R117, RPV Nozzle Ultrasonic Examination Summary Sheet
Report# R074, Examination Summary and Resolution Data Sheet
Visual Acuity Exam Record for Thomas Brown dated for 09/05/2009
Documentation of ASME Section XI, App. VII training for Thomas Brown dated 12/28/2009
Certificate of Method Qualification for Thomas Brown for Certification Period 9/15/2008 to 9/4/2012
Visual Acuity Exam Record for Marcie Kalkbrenner dated 8/8/2009
Qualification and Certification Summary for Marcie Kalkbrenner dated 1/20/2010

Section 1R11: Licensed Operator Requalification Program

Simulator Evaluation Guide OPL177.078, SRV Stuck Open, Earthquake, Loss of LPCI MG, Feedwater Line Break, Unisolable Steam Line Break on RCIC, Loss of High Pressure Makeup, Inability to Maintain RPV Water Level Above TAF, Emergency Depressurization (C1)
TRN-11.10, Annual Requalification Examination Development and Implementation, Rev. 15
Group 4 Crew Simulator Training Notebook

Section 1R12: Maintenance Effectiveness

SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting - 10CFR50.65, Rev. 9
0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting - 10CFR50.65, Rev. 34
Unit 1/2 and 3 Function 575-B, 575-C, & 575-E 4KV Power Supply and Busses System (a)(1) Plan, Rev 0
CDE 668, D 4KV Shutdown Board CASA Logic Relay Failure to Pick-Up
CDE 761, Failure of the 1A CS Pump to Trip
CDE 822, Stuck Relay Preventing Trip Signal To C 4KV Feeder Breaker
CDE 833, DC Control Power Lost to 3EA Shutdown Board
PER 203896, 3EA SD BD Loss of Control Power
PER 161766, CS Pump 1A Failure to Trip
PER 144272, 3B 4KV UB De-Energized When Attempting Transfer
PER 210828, System 575 (a)(1) Plan
PER 156416, Stuck Relay Preventing Trip Signal To C 4KV Feeder Breaker
Various PERs for Systems 202 and 211 for 2009

MREP Meeting Minutes dated 2/11/2009
MREP Meeting Minutes dated 3/30/2009
CDE 760, A3 EECW Pump Unavailability due to Shaft Knocking
CDE 849, A3 EECW Pump Unavailability due to Upper Shaft Replacement
CDE 850, A3 EECW Pump Unavailability due to Pump Replacement
CDE 851, A3 EECW Pump Unavailability due to Strainer Maintenance
PER 166464, A3 EECW Pump Knocking Noise
PER 161971, A3 EECW Pump Elevated Vibration
PER 204651, Maintenance Rule CDEs not initiated for EECW Pump Unavailability as Required
Unit 1/2 and 3 Function 071-B RCIC & 073-B & -C HPCI (a)(1) Plan, Rev 0
MREP Meeting Minutes dated 2/24/2009
CDE 810, U2 RCIC Failure to Inject
CDE 811, U3 RCIC Functional Failure due to Amphenol Connection Failure
CDE 840, U3 RCIC Failure due to Ribbon Cable Connection
PER 201649, During U3 RCIC Testing the EGR Connector Fell Out
PER 203537, Unit 2 RCIC Turbine Failed to Properly Start During Automatic Initiation
PER 208077, Unplanned entry into LCO
PER 216729, Maintenance Rule (a)(1) Plan for HPCI and RCIC Governor Control Systems

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

BFN Plant Risk and Protected Equipment Report for 1/29/2010
Unit 3 Sentinel report for 1/29/2010
PRA Evaluation Response BFN-0-10-015
0-TI-367, BFN Equipment to Plant Risk Matrix, Rev. 11
SPP-7.1, On-Line Work Management, Rev. 15
SPP-7.3, Work Activity Work Management Process, Rev. 4
NEDP-26, Probabilistic Risk Assessment (PRA), Rev. 1

SPP-9.11, Probabilistic Risk Assessment (PRA) Program, Rev. 0
BFN Plant Risk and Protected Equipment Report for 2/18/2010 and 2/19/2010
Unit 3 Sentinel report for 2/19/2010
PRA Evaluation Response BFN-0-10-030
SPP-7.2, Outage Management, Rev. 18
NEDP-26, Probabilistic Risk Assessment (PRA), Rev. 1
SPP-9.11, Probabilistic Risk Assessment (PRA) Program, Rev. 0
Unit 3 ORAM Safety Function Status reports for March 15, 16, 17 and 18, 2010
Control Room Operator Chronological Logs

Section 1R15: Operability Evaluations

3-SR-3.5.1.1(HPCI), Maintenance of Filled HPCI Discharge Piping, Rev. 3
EWR09MEB073033, Flow Rate and Time Required to Vent Gasses from the Highpoint HPCI Discharge Piping for Units 1, 2, and 3
EWR09MEB074031, Time Needed to Adequately Vent Lines in RHR
PER 171845, HPCI Gas Release
PER 172216, Unit 1 HPCI Timed Gas Release of 6 Minutes 43 Seconds
PER 174307,RHR System Venting
PER 211319, Unit 1 HPCI Timed Gas Release of 2 Minutes 58 Seconds
PER 201393, Unit 3 HPCI Timed Gas Release of 7 Minutes 5 Seconds
PER 208245, ECCS Venting Techniques
PER 209302, Unit 3 Core Spray Venting Technique
NEI 09-10, Guidelines for Effective Prevention and Management of System Gas Accumulation, Rev. 0
NRC Generic Letter 2008-01, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems
NRC Letter, William H. Ruland, dated May 28, 2009, Preliminary Assessment of Responses to Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems”, and Future Nuclear Regulatory Commission Staff Review Plans
NRC Letter, Eva A. Brown, dated August 24, 2009, Browns Ferry Nuclear Plant Units 1, 2, and 3 – Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems”, Request for Additional Information (TAC Nos. MD7799, MD7800, MD7801)
NRC Letter, Stewart N. Bailey, dated January 19, 2010, Browns Ferry Units 1, 2, and 3 – Closeout of Generic Letter 2008-01 “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems” (TAC Nos. MD7799, MD7800, MD7801)
NUREG-0927, Evaluation of Water Hammer Occurrence in Nuclear Power Plants, Rev. 1
TVA Letter, Michael A. Purcell, dated October 11, 2008, Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 – 9 Month Response to NRC Generic Letter (GL) 2008-01
TVA Letter, R. M. Krich, dated September 21, 2009, Response to Request for Additional Information for Regarding Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems”, Request for Additional Information (TAC Nos. MD7799, MD7800, MD7801)
PER 210961, UT of Unit 2 RHR II LPCI Injection and DW Spray
SR 110586, Deficiency With Past Operability Evaluation

PWROG FAI/08-70, Gas-Voids Pressure Pulsations Program, Rev. 0, dated 8/21/08
NEI Gas Intrusion Workshop-Diablo Canyon Void Evaluation
NRC Information Notices 87-10 and Supplement 1: Potential for Water Hammer During Restart of Residual Heat Removal Pumps, dated February 11, 1987 and May 5, 1997 respectively
NRC Information Notice 91-50 Supplement 1: Water Hammer Events Since 1991, dated July 17, 1997
NUREG-0927, Evaluation of Water Hammer Occurrence in Nuclear Power Plants, Rev.1
0-TI-360, Containment Leak Rate Programs, Rev. 28
0-TI-362, Inservice Testing of Pumps and Valves, Rev. 23
2-47E811-1, Flow Diagram Residual Heat Removal System, Rev. 66
2-47W452-220, NI-274-3R Isometric Torus Analysis of RHR System Pen X-210A&B, X-211A&B, Rev. 2
BFN-50-7074, General Design Criteria for Residual Heat Removal System, Rev. 19
Functional Evaluation for PER 210437, Rev. 0
Operational Decision-Making Issue Evaluation Document, PER 210437, Revs. 1 and 2
PER 210437, Gas Release During the Performance of 2-SR-3.5.1.1(RHR II)
PER 218493, Unit 2 Entry Into LCO 3.0.3 and Tech Spec Required Shutdown
PER 218875, RHR Design Criteria for Suppression Pool Cooling Mode Has Inconsistent Information
SR 111740, Late FE for PER 210437
Technical Requirements and Bases TR 3.3.3, ECCS Instrumentation, Rev. 14
Technical Requirements and Bases TR 3.5.4, Maintenance of Filled Discharge Pipe, Rev. 26
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3A EDG Operating Logs (3-OI-82, Illustration 2) for June 6, July 4, August 31, October 5, November 3, and November 29, 2009, and January 2, and January 30, 2010
3A EDG Operating Logs (3-OI-82, Illustration 2) for June 6, July 4, August 31, October 5, November 3, and November 29, 2009, and January 2, and January 30, 2010
3D EDG Operating Logs (3-OI-82, Illustration 2) for June 21, July 21, August 15&16, September 14, October 19, November14, and December 10, 2009, and January 16, and February 14, 2010

<u>Section 1R18: Plant Modifications</u>

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3-SI-3.2.4(DG C), EECW Check Valve Test on Diesel Generator C, Rev. 1, 2 and 3
3-SI-3.2.4(DG D), EECW Check Valve Test on Diesel Generator D, Rev. 1, 2 and 3
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WO 10540720, Implement TACF 3-10-002-067 on DG 3B
WO 10540753, Implement TACF 3-10-002-067 on DG 3C
WO 10540793, Implement TACF 3-10-002-067 on DG 3D
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PER 217065, Weaknesses Observed in Timely Step Performance Place-Keeping

PER 217313, I&C perform 3-SI-3.2.4(DG B) as requested by Operations
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3-OI-82, Standby Diesel Generator System, Section 5.5, Manual Slow Start at Diesel Generator Control Cabinet, Rev. 91
3-SR-3.8.1.1(3B), Diesel Generator 3B Monthly Operability Test, Rev. 36
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WO 08-724923-000, Replace Engine Cylinder Test Valves with New Kiene Test Valves

3-SR-3.5.1.6(CS I), Core Spray Flow Rate Loop I, Rev. 22
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WO 09-718461-000, Perform Molded Case Circuit Breaker Inspection and PM
WO 09-718468-000, Perform Molded Case Circuit Breaker Inspection and PM
3-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping (ASME Section III, Class I and II), Rev. 18
3-SI-4.7.A.2.G-3/71A, Primary Containment Local Leak Rate Test RCIC Turbine Steam Supply: Penetration X-10, Rev. 13
3-SR-3.6.1.3.5(RCIC), RCIC System MOV Operability, Rev. 30
3-SR-3.3.3.1.4(F), Verification of Remote Position Indicators for Reactor Core Isolation Cooling System Valves, Rev. 4
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MCI-0-000-ACT003, Maintenance of Limitorque Actuator Models SMB-00, Rev. 38
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SR 141715, Issues Identified by MOVATS Test on 3-FCV-71-2
WO 110729511, Repair RCIC Steam Line Inboard Isolation Valve
WO 09-720453-000, Connect and Disconnect Diesel Generator to the ADHR Alternate Feeder Breaker per ECI-0-72-ADR001
EPI-0-072-ADR001, Installation of Feeder Cables for Alternate Supply from Backup Diesel Generator for ADHR in Support of Refueling Outages
0-OI-72, Auxiliary Decay Heat Removal System
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PMT-0-000-MEC001, Leak Checks on Tube Fittings, Threaded, Flanged, Bolted or Welded Connections
WO 10559895, ADHR Secondary Pump Motor B Replacement (BFN-0-MTR-072-0272)
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WO 110707511, Repair/Replace breaker 0-72-251(A Secondary Pump Motor)
WO 110701172, Perform Inspection and Testing of ADHR Breakers
ECI-0-000-BKR008, Testing and Troubleshooting of Molded Case Circuit Breakers and Motor Starter Overload Relays
EPI-0-000-MCC001, Maintenance and Inspection of 480 VAC and 250 VDC Motor Control Centers
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WO 09-718598-000, Disassemble, Inspect, and Repair Testable Check Valve (BFN-2-CKV-075-026)
MCI-0-075-VLV004, Core Spray System Testable Check Valve CKV-75-26 and CKV-75-54 Disassembly, Inspection, Rework, and Reassembly
MCI-0-000-PCK001, General Maintenance Instructions for Valve Packing
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N-VT-4, System Pressure Test Visual Examination Procedure, Attachment 1, ASME Section XI VT-2 Visual Examination Report for BFN-2-CKV-075-0026
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2-SI-4.7.A.2.g-3/75a, Primary Containment Local Leak Rate Test Core Spray Injection: Penetration X-16A

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OTG-45, Refueling Activity Qualifications
SPP-10.8, Nuclear Fuel Management
0-GOI-100-2, New Fuel Operations
0-GOI-300-3C, Fuel Movement Operations During Refueling
0-GOI-100-3A, Refueling Operations (In-Vessel Operations), Attachment 4, Fuel/FRC Handler Certification
Training and certification record files for selected FHS

PER 220410, New Fuel Receipt Inspection and Fuel Handling Activities Conducted by SRO With Lapsed Qualifications
PER 220791, Inadequate LOR Training for FHS Relief Qualification Requirements
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3-SR-3.4.9.1(2), Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring during In-service Hydrostatic or Leak Testing
3-SR-3.4.9.1(2), Attachment 2, RPV Temperature Monitoring
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PER 224778, Inconsistencies between 3-SR-4.9.1(2), TS Bases and GE Evaluation
PER 223539, 3-SR-4.9.1(2) Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring Deficiencies
PER 222844, Potential Reactor Vessel Pressure/Temperature violation during Unit 3 Hydro
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3-SR-3.8.1.9(3B OL), Diesel Generator 3B Emergency Load Acceptance Test With Unit 3 Operating, Rev. 14
3-OI-82, Standby Diesel Generator System, Rev. 92
SR 134494, Use of Alligator Clips
PCR 10000390, 3-SR-3.8.1.9(3B OL) and other DG LATs Enhancements
3-SR-3.5.1.1(HPCI), Maintenance of Filled HPCI Discharge Piping. Rev. 3
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WO 110729511, Repair RCIC Steam Line Inboard Isolation Valve
2-SI-4.4.A.1 (Comp), Standby Liquid Control Comprehensive Pump Test, Rev. 1
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BFN Multibadge Tracking Form, Dosimeter 683325, 5/21/09
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10380056, U3R14 Drywell Shielding and Insulation Support (LHRA/Various Dress)
10380216, U3R14 Drywell Replace 3A/3B Recirc Motors (LHRA/Various Dress)
10380176, U3R14 DW CRD Change Out and Support (LHRA, Various Dress)
10380179, U3R14 DW CRD Change Out and Support (LHRA, Bubble Suit, Extremities)
10380097, U3R14 Drywell ISI, IWE, FAC (High Rad / Respirator)
09115036, U1 Power Reduction – Maintenance Activities (LHRA Various Dress)
09220808, Refuel Floor – Post U2C15 Outage Cleanup/Decon/Store Equipment (LHRA / Various Dress / Respirator)
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PER 206346, QA Audit SSA0906 – Rad Posting
PER 172081, Unanticipated dose rate alarm
PER 215502, Leaking Ni-63 sources
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PER 221481, Radworkers failing to recognize/obey HRA postings trend PER
PER 221112, Worker entered HRA (U2 SE Quad) without appropriate RWP/briefing
PER 219997, Nuclear security officer violated a HRA/CA boundary, U2 RXB 565’
PER 215769, Two FIN leak crew workers crossed a HRA boundary in the U2 HPCI quad
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NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 6
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Explanation of selected dosimeter alarms, spreadsheet, 3/23/10

<u>CAP Documents</u>
PER 209851, Several HRAs not secured in accordance with RCI-17
PER 213354, Inconsistency in GGMK controls for LHRA keys

<u>Section 4OA2: Identification and Resolution of Problems</u>
3-SI-4.5.C.1(2), EECW Pump Operation, Rev 109
3-SI-4.5.C.1(2-COMP), EECW Comprehensive Pump Test, Rev 10
0-SI-3.1.11, EECW Pump Baseline Data Acquisition and Evaluation, Rev. 26
0-SI-3.1.4, EECW Pump Performance, Rev. 49
0-TI-345, EECW Pump Curve Data Acquisition, Rev. 5
0-TI-362, Inservice Testing of Pumps and Valves, Rev. 23
0-TI-383, Evaluation of Test Results for the ASME OM Code Inservice Testing Program, Rev. 1
SPP-9.1, ASME Section XI, Rev. 8
PER 156818, RHRSW Pump B2 Failure
PER 175252, Quarterly Surveillance Results Used to Establish Baseline Flow Instead of Pump Curve Flow at the Reference Point
PER 175254, Temperature Affects Masking Pump Degradation in EECW IST Tests
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PER 213180, RHRSW/EECW IST Pump Test Non-Conformance
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Snubber Functional Test Scope all Sub-Groups for U3R14 RFO
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U3R14 Refueling Outage Snubbers database dated 3/23/10
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WO 09-718134-000, Remove, functionally test, and replace two BOP snubbers in accordance MPI-0-000-SNB004 and 0-SI-4.6.H-2A
0-SI-4.6.H-2A, Attachment 3 for BFN-3-SNUB-001-5041
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0-SI-4.6.H-2A, Attachment 4 for BFN-3-SNUB-001-5033
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PER 208627, Unit 2 HPCI Isolation during 2-SR-3.3.6.1.3(3DFT)
2-SR-3.3.6.1.3 (3DFT), HPCI Steam Line Space High Temperature Functional Test, Rev. 0
WO 09-724665-000, Trouble shoot and repair cause of HPCI isolation during HPCI Steam Line High Temperature Testing
WO 09-725842-001, Repair / replace connector and install grommet found missing during inspection of 1-LPNL-925-0229 connector.
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BFN Unit 2 Technical Specifications Section 3.5.1, ECCS - Operating
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BFN USFAR Section 7.3.4.7.10, Primary Containment Isolation System - Isolation Functions and Settings
BFN USFAR Section 7.3.4.8.10, Primary Containment Isolation System – Instrumentation
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PER 208627, Unit 3 HPCI Isolation during 2-SR-3.3.6.1.3(3DFT)
TACF 3-09-012-073, Isolation of Unit 3 HPCI 2" Turbine Exhaust Condensing Pot Drain Line
3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure, Rev. 52
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BFN USFAR Section 6.4, High Pressure Coolant Injection System
BFN USFAR Section 6.5, Safety Evaluation
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BFN-50-7073, HPCI Design Criteria, Rev. 19
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3-OI-73, High Pressure Coolant Injection System, Rev. 42
LER 05000260/2009-002, Leak In An ASME Code Class 1 Reactor Pressure Boundary Pipe, Rev. 1
PER 172551, Pressure Boundary Leak 2-SHV-74-49

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
ADS	Automatic Depressurization System
ARM	area radiation monitor
CAD	containment air dilution
CAP	corrective action program
CCW	condenser circulating water
CFR	*Code of Federal Regulations*
CoC	certificate of compliance
CRD	control rod drive
CS	core spray
DCN	design change notice
EECW	emergency equipment cooling water
EDG	emergency diesel generator
FE	functional evaluation
FPR	Fire Protection Report
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
LER	licensee event report
NCV	non-cited violation
NRC	U.S. Nuclear Regulatory Commission
ODCM	Off-Site Dose Calculation Manual
PER	problem evaluation report
PCIV	primary containment isolation valve
PI	performance indicator
RCE	Root Cause Evaluation
RCW	Raw Cooling Water
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RTP	rated thermal power
RPS	reactor protection system
RWP	radiation work permit
SDP	significance determination process
SBGT	standby gas treatment
SLC	standby liquid control
SNM	special nuclear material
SRV	safety relief valve
SSC	structure, system, or component
TI	Temporary Instruction
TIP	transverse in-core probe
TRM	Technical Requirements Manual
TS	Technical Specification(s)
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
WO	work order