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

July 30, 2009

Mr. Preston D. Swafford
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2009003, 05000260/2009003 AND 05000296/2009003

Dear Mr. Swafford:

On June 30, 2009, 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 July 6 and 17, 2009, 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 conduct augmented inspections on Unit 1, as required, in accordance with NRC letters dated May 16, 2007, December 6, 2007, and May 21, 2008. These Unit 1 augmented inspections were conducted to compensate for the lack of valid data for the Mitigating Systems Performance Index (MSPI) Performance Indicators (PIs). The augmented inspections are only considered to be an interim substitute for the invalid Unit 1 MSPI PIs until such time as complete and accurate PI data is developed and declared valid.

Based on the results of this inspection, one NRC-identified finding and three self-revealing findings of very low safety significance (Green) were identified. Three of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCV(s)) 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 Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at 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 Resident Inspector at 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 05000259/2009003, 05000260/2009003 and 05000296/2009003
w/Attachment: Supplemental Information

cc w/encl. (See page 3)

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Letter to Preston D. Swafford from Eugene F. Guthrie dated July 30, 2009

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2009003, 05000260/2009003 AND 05000296/2009003

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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.:	05000259/2009003, 05000260/2009003 and 05000296/2009003
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:	April 1, 2009 through June 30, 2009
Inspectors:	T. Ross, Senior Resident Inspector C. Stancil, Resident Inspector K. Korth, Resident Inspector R. Williams, Reactor Inspector (1R08) H. Gepford, Senior Health Physicist (2OS1, 2PS1, 4OA1, 4OA7) R. Hamilton, Senior Health Physicist (2OS2) W. Loo, Senior Health Physicist (2PS2)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000259/2009003, 05000260/2009003 and 05000296/2009003; 04/01/2009 – 06/30/2009; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Maintenance Risk Assessment, Access Control to Radiologically Significant Areas, and Event Followup.

The report covered a three month period of inspection by resident inspectors and reactor inspectors from the region. Three non-cited violations (NCV) and one Finding (FIN) 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. A Green self-revealing finding was identified for a failure to implement corrective actions in a timely manner to address excessive isophase bus cooling (IBC) system condensation that resulted in a main generator trip and Unit 1 reactor scram caused by water accumulation in the IBC ductwork, which created an electrical ground fault on the main generator isophase busses. An additional drain line was subsequently installed, and operating procedures were revised, to ensure any excessive condensation buildup would be removed from the IBC system during winter operation. This event was entered into the licensee's corrective action program as PER 163815.

This finding was determined to be greater than minor because it was associated with the Initiating Event Cornerstone attribute of Equipment Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was evaluated using Phase 1 of the At-Power Significance Determination Process, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting aspect of appropriate and timely corrective actions in the area of Problem Identification and Resolution because the license had identified an abnormal equipment condition related to excessive IBC system condensation for which specified corrective actions were not carried out (P.1.d). (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. A Green self-revealing noncited violation of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified for failure to adequately assess and manage shutdown risk associated with maintenance activities. Specifically, on May 2, 2009, the licensee performed a scheduled activity to install the 2A Recirculation Line nozzle plug in the reactor vessel. Installation of this plug isolated the common residual heat removal (RHR) system shutdown cooling (SDC) suction path while SDC availability was assumed in the shutdown risk assessment and had been designated as protected equipment as part of the specified risk management actions. Shortly after the nozzle plug was installed, the licensee recognized that SDC was no longer immediately available and promptly removed the plug to restore SDC availability. This event was entered into the licensee's corrective action program as problem evaluation report 170184.

This finding affected the Mitigating Systems cornerstone and was determined to be greater than minor because the licensee failed to effectively manage the prescribed significant compensatory measures (i.e., protection of Loop I RHR). The significance of this finding was evaluated using Inspection Manual Chapter (IMC 0609), Appendix G, “Shutdown Operations Significance Determination Process,” Table 1, “Losses of Control,” and Checklist 7 of Attachment 1, “BWR Refueling Operation with RCS Level > 23’ “, the inspectors determined that this finding was of very low safety significance (Green) because the event did not result a loss of control per Table 1 of Appendix G, or an actual loss of decay heat removal, and the SDC alignment was restored well within the time to boil. The cause of this finding was directly related to the cross cutting aspect of work activity coordination in the area of Human Performance, because the licensee failed to adequately evaluate the impact of the work and to communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant performance (H.3.b). (Section 1R13).

- Green. A Green non-cited violation of 10 CFR Part 50.65 (a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” was identified when the licensee failed to consider the impact of severe weather conditions on plant risk. Specifically, on May 1 and again on May 2, 2009, the licensee removed the A Emergency Diesel from service for planned maintenance during severe weather conditions (i.e., Tornado Warning and Tornado Watch, respectively) without re-evaluating the potential adverse affect upon the existing on-line risk assessment. The severe weather conditions only lasted about an hour each day. This issue was entered into the licensee's corrective action program as problem evaluation report 171402.

  The finding was determined to be greater than minor because the licensee’s risk assessment failed to consider unusual external conditions that were present or imminent (e.g., severe weather, offsite power instability). According to Inspection Manual Chapter 0609, Appendix K, Maintenance Risk Assessment and Risk Management Significance Determination Process, the significance of this finding was determined to be of very low safety significance (Green) based on an initiating events frequency of <1.0E-7, and a very low risk deficit due to the number of redundant emergency diesels and the short duration of the severe weather. The cause of this finding was directly related to the crosscutting aspect of complete and accurate procedures in the area of Human Performance because the licensee’s site-specific guidelines for assessing on-line risk did not require severe weather to be considered when determining plant risk nor did they require personnel to determine if severe weather is imminent prior to removing an emergency diesel generator from service (H.2.c). (Section 1R13).

Cornerstone: Occupational Radiation Safety

- Green A Green self-revealing non-cited violation of Technical Specifications 5.4.1, Procedures, was identified for a radiation worker who failed to follow the requirements of Radiation Work Permit 09270081 as required by procedure RCI-9.1, Radiation Work Permits, Rev. 57. The licensee has entered this issue into the corrective action program as problem evaluation report 171375.

  This finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process (Exposure Control) and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The finding was evaluated using the Occupational Radiation Safety Significance Determination Process and determined to be of very low safety significance (Green) because it was not related to As Low As Reasonably Achievable (ALARA) planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. The cause of this finding was directly related to the cross-cutting aspect of work practices in the area of Human Performance, because the radiation worker failed to use self-checking prior to passing through the swing gate into the posted high radiation area (H.4.a). (Section 2OS1)

B. Licensee Identified Violations

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and the corrective action program tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full rated thermal power (RTP) the entire report period except for three planned downpowers. On May 2, 2009, a planned downpower to approximately 90 percent RTP was conducted to perform a control rod adjustment. The unit was returned to full RTP the same day. On June 6, 2009, a planned downpower to approximately 70 percent RTP was conducted to execute a control rod sequence exchange. The unit was returned to full RTP the same day. On June 12, 2009, a planned downpower to approximately 75 percent RTP was conducted to repair Main Turbine Generator #3 Control Valve. The unit was returned to full RTP the same day.

Unit 2 began this report period at full RTP. On April 11, 2009, a planned downpower to 80 percent RTP was conducted to execute a control rod adjustment. The unit returned to full RTP on the same day. On April 25, 2009, the unit was shutdown for the Unit 2 Cycle 15 (U2C15) refueling outage. The unit was restarted June 8, 2009, but had to shutdown from approximately 15 percent RTP due to a sudden increase in unidentified reactor coolant system (RCS) leakage in excess of the Technical Specification limit of 2 gallons per minute in the previous 24 hours. Following repairs, the reactor was restarted on June 15, 2009. The unit was shutdown again from approximately 14 percent RTP on June 16, 2009 due to failure of the same MSRV to properly lift and reseat during testing. After the MSRV was replaced, Unit 2 was restarted on June 20, 2009. The unit achieved full RTP power on June 27, 2009 and remained at full RTP for the remainder of the report period.

Unit 3 operated at full RTP the entire report period except for eight planned downpowers. Four of the power reductions were conducted to between 90 and 95 percent RTP to repair reactor feedwater heater or moisture separator normal level control valves. These downpowers were conducted on April 3, May 22, May 29, and June 23, 2009, and in each case, power was restored to full RTP on the same day. On April 4, 2009, a downpower was conducted to 75 percent RTP for a control rod adjustment. On April 15, 2009, a downpower to approximately 58 percent RTP was conducted to replace two power cells in the 3B Variable Frequency Drive. The unit was returned to full RTP on April 17, 2009. On May 21, 2009, a planned downpower to approximately 95 percent RTP was conducted to recover a control rod that had been inserted for maintenance, and promptly returned to full RTP. Then on June 27, power was reduced to 75 percent for a rod sequence exchange and main condenser cleaning. The unit was returned to full RTP the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

Prior to and during the onset of hot weather conditions, the inspectors reviewed the licensee's implementation of 0-GOI-200-3, Hot Weather Inspection, including

Attachment 1, Hot Weather Operational Checklist. The inspectors also reviewed the Hot Weather Discrepancy Log (PA-104); and discussed implementation of 0-GOI-200-3 with responsible Work Control and Operations personnel and management. Furthermore, the inspectors conducted walkdowns of potentially affected risk significant equipment systems relied on to cool the 480v and 4Kv Shutdown Board Rooms.

b. Findings

No findings of significance were identified.

.2 Offsite and Alternate AC Power Systems Readiness

a. Inspection Scope

Prior to the summer season, inspectors reviewed electrical power design features, onsite risk and work management procedures, and corporate transmission and power supply procedures to verify appropriate operational oversight and assurance of continued availability of offsite and alternate AC power systems. Inspectors verified that communications protocols existed between the transmission system operator and Browns Ferry Nuclear Plant for coordination of off-normal and emergency events affecting the plant, event details, estimates of return-to-service times, and notifications of grid status changes. Inspectors also verified that procedures included controls to adequately monitor both offsite AC power systems (including post-trip voltages) and onsite alternate AC power systems for availability and reliability. Furthermore, inspectors interviewed onsite licensed operators and offsite transmission personnel to determine their understanding and implementation of the power monitoring and assessment process. Inspectors reviewed the material condition of offsite AC power systems and onsite alternate AC power systems to the plant, including switchyard and transformers. This review included review of outstanding work orders affecting these systems and a walkdown of the switchyard with operations personnel to ensure the systems will continue to provide appropriate "as designed" capabilities.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors conducted three equipment partial alignment walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, with 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.

- Unit 2 Core Spray (CS) System, Division II
- Unit 3 High Pressure Coolant Injection (HPCI) System
- Unit 3 Residual Heat Removal (RHR) System, Division II

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, was in place.

- Unit 1 Reactor Building Elev. 621, 1A Electrical Board Room (EBR) (FA-5)
- Unit 1 Reactor Building Elev. 621, Shutdown Board Room 1A (FA-6)
- Unit 1 Reactor Building Elev. 621, Shutdown Board Room 1B (FA-7)
- Unit 2 Control Building Elev. 593, Battery Board Room 2 and Battery Room 2 (FA-18)
- Unit 2 Reactor Building Elev. 639 South of Column Line R (FZ 2-6)
- Unit 3 Reactor Building Elev. 593, 480v RMOV Board Room 3B (FA-12)
- Unit 3 Reactor Building Elevs. 621 and 639 North of Column Line R (FZ 3-4)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08G)

.1 Non-Destructive Examination (NDE) Activities and Welding Activities

a. Inspection Scope

From May 4 to May 8, 2009, the inspectors 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 2, 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 are outside the scope of ASME Section XI requirements which are 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:

- Liquid penetrant (PT) and visual exam (VT-3) of component 2-47B408S0043 and its integrated attachment, ASME Class 1 Category B-K, welded attachment
- Phased array ultrasonic testing (UT) examination of weld GFW-2-32, ASME Class 1 Category B-J, Tee to pipe weld

The inspectors reviewed the following examination from the previous outage with a relevant indication that was analytically evaluated and accepted for continued service to verify that the acceptance was in accordance with ASME Code:

- NOI #U2C14-049 – N9 nozzle to vessel weld

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 #09-711929-000 – Welded replacement of valve BFR-1-SHV-067-0655

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Internal Inspections

a. Inspection Scope

The inspectors reviewed the following NDE activities associated with the inspection of reactor vessel internal components per the BWRVIP:

- EVT-1 of Jet Pump restrainer assemblies including the set screws and wedge assemblies
- VT-3 of feed water sparger pins

The inspectors also 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. The inspectors reviewed these corrective action documents to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The 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 corrective action documents reviewed by the inspectors are listed in the report attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On April 6, 2009, the inspectors observed licensed operator requalification simulator examination for two crews. Each crew received the same examination scenario - "Group 6 Isolation and Main Steam Line Break in Containment".

The inspectors specifically evaluated the following attributes related to each operating crew's 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 a 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.

1R12 Maintenance Effectiveness

.1 Routine

a. Inspection Scope

The inspectors reviewed two specific equipment issues listed below for structures, systems and components (SSC) within the scope of the Maintenance Rule (MR) pursuant to 10CFR50.65 with regard to some or all of the following attributes: (1) Work practices; (2) Identifying and addressing common cause failures; (3) Scoping in accordance with 10 CFR 50.65(b); (4) Characterizing reliability issues (e.g., functional failures); (5) Trending key parameters for condition monitoring; (6) Tracking SSC unavailability; (7) Appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2); (8) System classification in accordance with 10 CFR 50.65(a)(1); and (9) 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. The inspectors also reviewed, as applicable, work orders, surveillance records, problem evaluation reports (PERs), system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met.

- Unit 2 HPCI System Unavailability Exceeded MR Performance Criteria.
- Unit 1 Reactor Core Isolation Cooling (RCIC) System Failures

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For planned online work and/or emergent work that affected the combinations of risk significant systems listed below, the inspectors reviewed six 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-7.0, Work Management; SPP-7.1, On-Line Work Management; 0-TI-367, BFN Equipment to Plant Risk Matrix; and BP-336, Risk Determination And Risk Management. The inspectors also evaluated the adequacy of the licensee's risk assessments and implementation of RMAs.

- Unit 2 CS Loop I, 2A Control Rod Drive (CRD) Pump, A Control Bay Chiller, 1A EBR Air Handling Unit, and 2A Condenser Circulating Water (CCW) Pump out of service (OOS)
- Unit 2 Loop I CS, 2A CRD pump, A1 Residual Heat Removal Service Water (RHRSW) pump, 2C CCW pump, 2C Condensate Booster pump, and 2C Reactor Feedwater pump OOS
- Unit 1 Loop II Core Spray, 1B CRD pump (aligned to U2), A3 and C3 Emergency Equipment Cooling Water (EECW) pumps, and 2-RHR-I crosstie OOS
- Shutdown Risk with Secondary System Pump B of Alternate Decay Heat Removal (ADHR) System OOS and Shutdown Cooling not immediately available due to installation of recirculation line nozzle plug
- Unit 1 and 2 A Emergency Diesel Generator (EDG) during Severe Weather
- Unit 2 Unplanned Orange Shutdown Risk for ADHR B-Train, Shutdown Cooling, Loop II RHR, and Loop I and II CS OOS

b. Findings

The inspectors identified two findings involving maintenance risk assessments, as documented below.

.1 Inadequate Shutdown Risk Assessment During Outage Maintenance

Introduction: A self-revealing non-cited violation (NCV) of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified for failure to adequately assess and manage shutdown risk associated with outage maintenance activities. Specifically, on May 2, 2009, the licensee performed a scheduled activity to install the 2A Reactor Recirculation suction line nozzle plug in the reactor vessel. This plug isolated the common shutdown cooling (SDC) suction path while SDC was assumed to be available in the existing outage risk assessment and had been designated as protected equipment .

Description: On May 1, 2009, Unit 2 was in a refueling outage in Mode 5 with greater than 23' of water above the fuel and a time-to-boil of approximately 24 hours. The B secondary side alternate decay heat removal (ADHR) pump was out of service for unplanned maintenance. In accordance with SPP 7.2, Outage Management, a shutdown risk assessment was conducted using Outage Risk Assessment Management (ORAM) software. In order to maintain the unit in a Yellow outage risk condition for the decay heat removal safety function, the Loop I RHR system for SDC was being protected (i.e., risk management action) as a redundant system for decay heat removal. The ADHR system was in service using the A secondary side pump, SDC was secured but still available.

As part of the outage schedule, the 2A Reactor Recirculation suction line was to be plugged to perform outage maintenance. The predecessor activity to the reactor recirculation suction line nozzle plug installation was to ensure SDC was out of service. This activity was incorrectly reported as complete when RHR pumps were secured, and SDC was placed in standby, at which point the outage control center (OCC) directed Maintenance to install the plug. Multiple personnel in different organizations knew that the plug was being installed. However, it was not recognized and/or communicated that this action would isolate both the reactor water cleanup (RWCU) and SDC suction path, and thereby make the designated protected equipment unavailable for use. Shortly after the nozzle plug was installed, at 0016 on May 2, 2009, the Unit 2 control room received a RWCU Recirculation Pump Flow Low alarm and operators responded per the Alarm Response Procedure (ARP). Attempts were made to raise the flow per the ARP and when there was no response the 2A RWCU pump was secured. At 0030, the Main Control Room (MCR) staff was informed that the 2A Recirculation Line suction plug had been installed. Control room personnel promptly recognized that installation of this plug had resulted in the RWCU low flow condition and had effectively isolated shutdown cooling. Maintenance personnel were immediately directed to remove the plug. The nozzle plug was reported to be removed from the recirculation line at 0620 on May 2, 2009.

Analysis: The inspectors determined that the risk management actions put in place pursuant to 10 CFR 50.65(a)(4) for in progress outage maintenance activities were not effectively managed to prevent disabling the protected SDC system and constituted a performance deficiency. This finding affected the Mitigating Systems cornerstone and was determined to be greater than minor according to Inspection Manual Chapter (IMC) 0612, Appendix B, Issue Screening, because the licensee failed to effectively manage the prescribed significant compensatory measures (i.e. protection of Loop I RHR). The significance of this finding was evaluated using IMC 0609, Appendix K, Maintenance Risk Assessment and Risk Management Significance Determination Process, which stated that if the finding involved maintenance activities during shutdown conditions, then the appropriate checklist reflecting the plant shutdown mode from IMC 0609, Appendix G, Attachment 1, should be checked to determine risk significance. According to IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” Table 1, “Losses of Control,” and Checklist 7 of Attachment 1, “BWR Refueling Operation with RCS Level > 23’” , the inspectors determined that this finding was of very low safety significance (Green) because the event did not result a loss of control per Table 1 of MC0609, Appendix G, or an actual loss of decay heat removal and the SDC alignment was restored well within the time to boil. As such, Appendix G did not require the significance of the finding to be quantified by a Phase 2 or 3 analysis. During the event, ADHR was in service removing reactor decay heat, and there was adequate time to restore SDC to service (i.e., remove the plug) before boiling of the RCS would occur if ADHR were lost (i.e., time to boil about 24 hours). The RHR SDC function was restored in a relatively short period of time (approximately 6 hours). The cause of this finding was directly related to the cross cutting aspect of work activity coordination in the area of Human Performance, because the licensee failed to adequately evaluate the impact of the work and to communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant performance. Specifically, the OCC authorized installation of the recirculation line plug even though the plant was relying on SDC availability as defense in depth for their shutdown risk assessment (H.3.b).

Enforcement: 10 CFR 50.65(a)(4) required, in part, that prior to performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this, on May 2, 2009, the licensee failed to adequately assess and manage shutdown risk associated with outage maintenance activities related to SDC availability to provide for the decay heat removal safety function. Because the finding was determined to be of very low safety significance and has been entered into the licensee's CAP as PER 170184, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000260/2009003-01, "Failure to Assess and Manage Shutdown Risk Associated With Outage Maintenance Activities."

.2 Failure to Perform an Adequate Risk Assessment during Severe Weather Conditions

Introduction: The inspectors identified a Green NCV of 10 CFR 50.65(a)(4) involving the licensee's failure to consider the impact of severe weather on plant risk.

Description: At 1550, on May 1, 2009, with Unit 1 at 100 percent power, the licensee removed the A EDG from service for planned maintenance while the A3 EECW pump and the RHR crosstie were already OOS. At 1611, a Tornado Warning was issued by the National Weather Service. However, the licensee failed to re-evaluate their risk assessment prior to, or during the onset of, severe weather conditions to consider the potential impact of severe weather on plant risk. The Tornado Warning expired at 1715. Then on May 2, 2009, the A EDG was declared inoperable for surveillance testing at 1503. At 1510, a Tornado Watch was issued by the National Weather Service. The licensee aborted the diesel surveillance and restored it to service, but severe weather conditions were not considered in the risk assessment either before or afterwards.

Section 3.2.A, of Standards Programs and Processes (SPP) 7.1, On-Line Work Management, required that a risk assessment of scheduled on-line maintenance be performed prior to implementation, and emergent work was to be evaluated against the assessed scope. According to section 3.2.B.1.n of SPP 7.1, this risk assessment was required to include external event considerations involving the potential impacts of weather or other external conditions relative to the proposed maintenance evolution if these external impacts (e.g., weather, external flooding, grid reliability, and other external impacts) were imminent or have a high probability of occurring during the planned out of service duration. Furthermore, section 3.2.2 of SPP 7.1, required that the assessment of risk for scheduled activities be re-performed if there were significant changes in external conditions (e.g., weather or offsite power availability). The licensee's failure to consider severe weather conditions as part of their on-line risk assessments on May 1 and 2, 2009, were contrary to the provisions of SPP-7.1. The sub-tier instructions used by the licensee for assessing on-line risk at Browns Ferry included BP-336, Risk Determination and Risk Management, and 0-TI-367, BFN Equipment to Plant Risk Matrix. These Browns Ferry site specific guidelines did not include consideration of severe weather impact on on-line risk.

Analysis: The failure of the licensee to consider the impact of severe weather on plant risk was a performance deficiency which could have led to underestimating the risk of plant maintenance activities and/or failure to take actions to minimize that risk. This finding was determined to be greater than minor according to IMC 0612, Appendix B, Issue Screening, because the licensee's risk assessment failed to consider unusual external conditions that were present or imminent (e.g., severe weather, offsite power instability). The significance of this finding was determined to be of very low safety significance (Green) according to IMC 0609, Appendix K, Maintenance Risk Assessment and Risk Management Significance Determination Process, based on an initiating events frequency of <1.0E-7, and a very low risk deficit due to the number of redundant emergency diesels and the short duration of the severe weather. This issue was entered into the licensee's corrective action program as PER 171402.

The cause of this finding was directly related to the crosscutting aspect of complete and accurate procedures in the area of Human Performance because the licensee's site specific guidelines for assessing on-line risk did not require severe weather to be considered when determining plant risk nor did they require personnel to determine if severe weather is imminent prior to removing an emergency diesel generator from service (H.2.c).

Enforcement: 10 CFR 50.65 (a)(4), required in part, that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this, on May 1 and 2, 2009, plant personnel failed to assess and manage risk for maintenance activities during unusual external conditions (i.e., severe weather). Because this finding was determined to be of very safety risk significance (Green) and has been entered into the licensee's CAP as PER 171402, this violation is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy: NCV 05000259/2009003-02, "Failure to Perform an Adequate Risk Assessment during Severe Weather Conditions."

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the five 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 NEDP-22, Functional Evaluations and PIDP-3, Operability and Reportability Reviews of PERs, 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.

- Unit 2 Core Spray Room Cooler Low Flow/Differential Pressure (PER 167344)
- Unit 1 Turning Gear Oil Pump Failure to Load Shed from 480V Board (PER 166870)
- Unit 3 Main Steam Ruggedness Boundary/Alternate Leakage Treatment Path (3-SHV-1-743 Packing Leak in 3B Steam Jet Air Ejector Room) (PER 147819)
- Unit 1 Narrow Range Suppression Pool Level Indicators in Divergence (PER 169830)
- Unit 2 Degraded Primary Containment Coatings (PER 171208)

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed one temporary modification listed below to verify regulatory requirements were met, along with procedures such as 0-TI-405, Plant Modifications and Design Change Control; 0-TI-410, Design Change Control; and SPP-9.5, Temporary Alterations. The inspectors also reviewed existing licensee documentation 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.

- PER 168128, Vibration Instrumentation for Monitoring Unit 3 RHRSW HX Outlet Valve Vibrations

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the Design Change Notice (DCN) and completed work packages (WO 08-724013-018 and 08-724013-21) for DCN 69528, Replace ASCO Solenoid Valves with Similar AVCO Valves, including related documents and procedures. The inspectors reviewed licensee procedures 0-TI-405, Plant Modifications and Design Change Control, and SPP-9.3, Plant Modifications and Engineering Change Control, and observed part of the licensee's activities to implement this design change made while the unit was online. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify that the modifications had not affected system operability/availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the six post-maintenance tests (PMT) listed below to verify that procedures and test activities confirmed SSC operability and functional capability following maintenance. The inspectors 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 also witnessed the test and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors verified that PMT activities were conducted in accordance with applicable WO instructions, or procedural requirements, including SPP-6.3, Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors reviewed problems associated with PMTs that were identified and entered into the CAP.

- Common: PMT for replacement of A3 EECW Pump, in accordance with 3-SI-4.5.C.1(2), EECW Pump Operation; 0-SI-4.5.C.1(4), EECW Annual Flow Rate Test; 0-SI-3.1.11, EECW Pump Baseline Data Acquisition and Evaluation; and Work Order 09-715933-003, Remove A3 EECW Pump for Inspection
- Unit 2: PMT for Control Rod Drive Hydraulic Control Unit Insert Riser Isolation Valve, 2-ISV-085-612/3455, per WO 08-712349-000 and Maintenance Procedures MCI-0-000-GTV001, General Maintenance Instructions for Gate Valves; MCI-0-000-PCK001, General Maintenance Instructions for Valve Packing; and MSI-0-000-PLG001, Installation of Freeze Seals.
- Unit 1: PMT for Unit 2 North Header Supply to Northeast Core Spray Room Cooler Valve, 1-SHV-067-0655, per WO 09-711929-000; 0-OI-67, Emergency Equipment Cooling Water System; and 1-SI-3.2.4, EECW Check Valve Test
- Unit 2: PMT for replacement of the motor on the Core Spray Pump 2D, in accordance with 2-SR-3.5.1.6(CS II-Comp), Core Spray Loop II Comprehensive Pump Test and Work Orders 07-721562-002, -003, -005 and -007.
- Unit 1: PMT for Modification/Replacement of Unit 2 Loop II Residual Heat Removal Service Water Heat Exchanger Outlet Valves, 2-FCV-023-46 and -52, per WOs 07-722378-008, -012, and -014, and procedures ECI-0-000-MOV009, Testing of Motor Operated Valves Using MOVATS Universal Diagnostic System (UDS) and Viper 20; 2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test; MCI-0-023-VLV001, RHR Service Water Motor Operated Valves 1/2-FCV-23-34, -40, -46, and 52 Disassembly, Inspection, Rework, and Reassembly; and MCI-0-000-PCK001, Generic Maintenance Instruction for Valve Packing.
- Unit 2: PMT for Replacement of Loop II RHR Pump Motors in accordance with 2-SR-3.5.1.6 (RHR II-COMP), RHR Loop II Comprehensive Pump Test, Work Orders 07-721570-002, -005, and -006 and Work Orders 07-721571-002, -005, and -007.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 2 Scheduled Refueling Outage (U2C15)

a. Inspection Scope

During April 25 to June 26, 2009, the inspectors examined critical outage activities to verify that they were conducted in accordance with TS, applicable procedures, and the licensee's outage risk assessment and management plans through the end of the reporting period. Some of the more significant inspection activities conducted by the inspectors were as follows:

Outage Risk Assessment

Prior to the Unit 2 Cycle 15 (U2C15) refueling outage that began on April 25, 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 U2C15 Refueling Outage Reports, including the ORAM Safety Function Status, and regularly attended the twice a day outage status 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 2 in accordance with licensee procedures SPP-12.1, Conduct of Operations; 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; and 2-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 2-OI-74, Residual Heat Removal System (RHR); 2-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Alternate Decay Heat Removal System Failures; and conducted a main control room panel and in-plant walkdowns of system 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., Equipment Clearance Order 0-TO-2009-0001, Clearance 0-067-0010, for Cutout and Replacement of 1-SHV-67-655, North Header Supply to NE Core Spray Room Cooler).
- Verified RCS inventory controls, especially during evolutions involving operations with the potential to drain the reactor vessel (OPDRV) controlled per 2-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 MCR, reactor 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 2-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, on numerous occasions, the inspectors witnessed fuel handling operations during the three Unit 2 reactor core fuel shuffles performed in accordance with TS and applicable operating procedures, such as GOI-100-3A, Refueling Operations (In Vessel), GOI-100-3B, Operations in the Spent Fuel Pool, and GOI-100-3C, Fuel Movement Operations During Refueling. The inspectors verified specific fuel movements as delineated by the Fuel Assembly Transfer Sheets (FATF).

Torus and Drywell Closeout

On May 30, the inspectors reviewed the licensee's final closure of the Unit 2 suppression pool in accordance with 2-GOI-200-2, Primary Containment Initial Entry and Closeout, and performed an independent detailed closeout inspection of the Unit 2 torus.

The inspectors conducted several independent detailed closeout inspections of the Unit 2 drywell prior to and ending on June 19. The inspectors also reviewed and verified the licensee's conduct of 2-GOI-200-2.

Restart Activities

The inspectors specifically conducted the following:

- Attended the restart Plant Oversight Review Committee meetings

- Witnessed heatup and pressurization of Unit 2 reactor pressure vessel in accordance with 2-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor pressure Vessel and Associated Piping

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

- Reviewed 2-SR-3.6.1.1.1(OPT-A) Primary Containment Total Leak Rate - Option A, Revision 6

- Witnessed Unit 2 criticality and/or power ascension per 2-GOI-100-1A, Unit Startup and Power Operation, for the startups on June 8, 15, and 20. Also witnessed the Unit 2 shutdowns on June 11 and 16.

- Witnessed and reviewed Unit 2 RCS heatup and pressurization per 2-SR-3.4.9.1, Reactor Heatup and Cooldown Rate Monitoring, for the startups conducted on June 8, 15, and 20. Also witnessed the subsequent RCS cooldowns following the shutdowns on June 11 and 16.

Corrective Action Program

The inspectors reviewed PERs generated during U2C15 and 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 corrective actions of several PERs were also reviewed for completeness.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions and/or reviewed completed test data for the following five 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:

- 1-SR-3.5.1.6(CS I), Quarterly CS System Flow Rate Test Loop I

Routine Surveillance Tests:

- 0-SR-3.8.1.9(D), Diesel Generator D Emergency Unit 2 Load Acceptance Test
- 2-SR-3.1.4.1, Scram Insertion Times
- 2-SR-3.4.3.2, Main Steam Relief Valve Manual Cycle Test

Containment Isolation Valve Tests:

- 2-SI-4.7.A.2.a-f, Primary Containment Integrated Leak Rate Test

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

During the report period, the inspectors observed an Emergency Preparedness (EP) drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures on April 7, 2009, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room and certain Emergency Response Facilities to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure and other applicable Emergency Plan Implementing Procedures. 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)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Control: The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress associated with the U2C15 refueling outage. The inspectors directly observed implementation of administrative and physical radiological controls; evaluated radiation worker (radworker) and radiation protection technician (RPT) knowledge of and proficiency in implementing radiation protection requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation areas, high radiation areas (HRAs), and potential airborne radioactivity areas established within the radiation control area (RCA) of the Unit 2 (U2) drywell, Unit 1 (U1), U2, and Unit 3 (U3) reactor buildings, U1/U2/U3 turbine building, and radioactive waste (radwaste) processing and storage locations. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. Results were compared to current licensee surveys and assessed against established postings and Radiation Work Permit (RWP) controls. Licensee key control and access barrier effectiveness were evaluated for selected U1, U2, and U3

Locked High Radiation Area (LHRA) and Very High Radiation Area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with radiation protection (RP) supervisors.

Controls and their implementation for storage of irradiated material within the spent fuel pool (SFP) were reviewed and discussed in detail. Established radiological controls were evaluated for selected U2C15 outage tasks including transverse in-core probe under-vessel work, control rod drive accumulator maintenance, diving activities associated with steam dryer work, in-service inspection, scaffolding support, insulation, shielding, motor replacements In addition, licensee controls for areas where dose rates could change significantly as a result of plant shutdown and refueling operations were reviewed and discussed.

For selected tasks, the inspectors attended pre-job briefings and reviewed RWP details to assess communication of radiological control requirements to workers. Occupational workers' adherence to selected RWPs and RPT proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Electronic dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results for drywell and refueling floor activities.

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. Worker exposure as measured by ED and by licensee evaluations of skin doses resulting from discrete radioactive particle or dispersed skin contamination events during current U2C15 outage activities were reviewed and assessed. For HRA tasks involving significant dose rate gradients, e.g. steam dryer repair activities by divers in the fuel pool, 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 2008 and the current U2 refueling outage.

The inspectors walked-down the Independent Spent Fuel Storage Installation (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 RP staff. Environmental monitoring results for direct radiation from the ISFSI were reviewed and inspectors observed the placement and physical condition of thermoluminescent dosimeters (TLDs) around the facility.

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 2OS1 of the report Attachment.

<u>Problem Identification and Resolution:</u> Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. This included review of selected PERs related to radworker

and RPT 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. 15. 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 2OS1 of the report Attachment.

The inspectors completed the 21 required line-item samples described in Inspection Procedure (IP) 71121.01. The inspectors also completed the radiation protection line-item sample activities specified in IP 60855.1.

b. Findings

Introduction: A self-revealing Green NCV of TS 5.4.1, Procedures, was identified for a failure to comply with the requirements of Radiation Work Permit (RWP) 09270081, as required by procedure RCI-9.1, Radiation Work Permits.

Description: On May 15, 2009, a machinist received a briefing on RWP 09270081, U2C15 Reactor Building Movats (Various Dress), with an electronic dosimeter (ED) dose alarm setpoint of 50 millirem (mrem) and a dose rate setpoint of 80 millirem per hour (mrem/hr). Based on the work location communicated to RP by the machinist, he was briefed on the radiological conditions in the HPCI room, located adjacent to the U2 SE Quad on the 519' elevation. The anticipated dose rates in the HPCI room were < 1 mrem/hr general area, with some localized areas < 5 mrem/hr. The RWP required the worker to review appropriate survey data prior to entry and did not allow entry into high radiation areas.

The worker then proceeded to the U2 519' SE Quad and, instead of entering the HPCI room, passed through a high radiation area (HRA) swing gate and entered a posted HRA in order to look for the valve he was tasked to work on, located near the RHR pumps. Shortly after entering the HRA, the worker received an unanticipated dose rate alarm of 84 mrem/hr, identified when he logged-off the RWP. Because the worker did not hear the ED alarm, he continued to work in the posted high radiation area, unaware of the radiological conditions because they had not been covered during his briefing. Actual radiological conditions were 30-40 mrem/hr general area, and 50 mrem/hr at 30 cm from the RHR pumps. Based on the briefed radiological conditions, it was expected that the worker would receive 1-2 mrem to perform the valve work; because he performed the work in a significantly higher dose rate field than had been briefed, he received a total dose of 19 mrem.

The machinist had successfully completed radworker training, which detailed the different types of radiological postings/barriers and requirements for entry into areas such as HRAs. Because a swing gate served as a physical barrier to inadvertent entry into the HRA on the U2 519' SE Quad, the worker missed an opportunity to self-check his actions prior to entry into the area.

Analysis: The inspectors determined that the failure to follow the requirements of RWP 09270081 was a performance deficiency. This finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process (Exposure Control) and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation.

The finding was evaluated using the Occupational Radiation Safety SDP and determined to be of very low safety significance (Green) because it was not related to As Low As Reasonably Achievable (ALARA) planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. The cause of this finding was directly related to the cross-cutting aspect of work practices in the Human Performance area, because the machinist failed to use self-checking prior to passing through the swing gate into the posted high radiation area (H.4.a).

Enforcement: Technical Specifications 5.4.1, Procedures, required the licensee to implement the procedures contained in Regulatory Guide (RG) 1.33, Rev. 2, Appendix A. RG 1.33, Appendix A, Section 7.e. which required written procedures for Radiation Protection, including access control to radiation areas and a radiation work permit system. Licensee procedure RCI-9.1, Radiation Work Permits, Rev. 57, required radworkers to use the appropriate RWP for entries into the RCA and to comply with all RWP special instructions. RWP 09270081, U2C15 Reactor Building Movats (Various Dress), Worker Instruction Number (No.) 3 required workers to review appropriate survey data prior to entry and Worker Instruction No. 4 specified no entry into high radiation areas on this RWP. Contrary to this, on May 15, 2009, a machinist passed through a swing gate and entered a posted high radiation area on RWP 09270081 without having reviewed appropriate survey data for the area, contrary to the RWP Worker Instructions Nos. 3 and 4. Because this finding was determined to be of very low safety significance and has been entered into the licensee's CAP (PER 171375), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000259,260,296/2009003-03, "Failure to Comply with the Requirements of an RWP by Entering a Posted High Radiation Area."

2OS2 As Low As Reasonably Achievable Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable: The inspectors evaluated ALARA program guidance and implementation for ongoing tasks associated with the U2C15 refueling outage. In addition, the inspectors evaluated the post-outage ALARA activities associated with the Unit 1 Cycle 7 (U1C7) refueling outage. The inspectors reviewed and discussed with licensee staff various ALARA work plan documents, including dose estimates and prescribed ALARA controls for selected outage work activities expected to incur significant collective doses. The inspectors reviewed the implementation of dose-reduction initiatives for high person-rem expenditure tasks. The inspectors evaluated these elements of the ALARA program for consistency with the methods and practices delineated in applicable licensee procedures.

The inspectors evaluated the implementation and effectiveness of ALARA planning and program initiatives during work in progress. The inspectors made direct field observations of U2 work activities involving the steam dryer repair, specifically diving operations including diver preparation, supervision, dosimetry placement and removal, and work area radiation surveys. The inspectors observed work on the CRD accumulators, scaffold erection, and in-service inspection. The inspectors interviewed radiation workers and RPT staff to assess their understanding of dose reduction initiatives and their current and expected final accumulated occupational doses at completion of the task.

Projected RWP dose expenditure estimates from U2C15 and U1C7 refueling outage efforts were compared to actual dose expenditures, and noted differences were discussed with cognizant ALARA staff. Changes to dose budgets relative to changes in job scope were identified and discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel.

The inspectors evaluated the implementation and effectiveness of selected program initiatives with respect to source-term reduction. The inspectors reviewed source term reduction plan that would duplicate many of the source term reduction efforts that have been applied to Unit 1 for the other two units. The inspectors noted that many items in the source term reduction plan had been eliminated or deferred past calendar year (CY) 2014 into the next 5-year planning cycle. The inspectors discussed the potential impacts of the eliminations and deferrals with RP management. The effectiveness of selected shielding packages installed for the current outage was assessed through completion of independent radiation surveys and comparison to applicable licensee survey records and expected planning data.

The plant collective exposure histories for CY 2005, 2006 and 2007, taken from data reported to the NRC pursuant to 10 CFR 20.2206(c), were reviewed and discussed with licensee staff, as were established goals for reducing collective exposure. The inspectors reviewed the applicable guidance and examined dose records of declared pregnant workers during CY 2007 and 2008 to evaluate current gestation doses for declared pregnant workers.

ALARA activities were evaluated against the requirements specified in 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. In addition, licensee performance was evaluated against RG 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable, and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Sections 2OS1 and 2OS2 of the report Attachment.

<u>Problem Identification and Resolution</u>: The inspectors reviewed the licensee corrective action documents associated with ALARA activities. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with the corrective action program. Specific self-assessments, audits, and PERs reviewed and evaluated in detail for this inspection area are identified in Section 2OS2 of the report Attachment.

The inspectors completed the 15 required line-item samples and 14 optional line-item samples, for a total of 29 line-item samples, as described in IP 71121.02.

b. <u>Findings</u>

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 <u>Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems</u>

a. Inspection Scope

Groundwater Monitoring: The inspectors discussed current and future programs for onsite groundwater monitoring with licensee staff, including number and placement of monitoring wells and identification of plant systems with the most potential for contaminated leakage. The inspectors also reviewed procedural guidance for identifying and assessing onsite spills and leaks of contaminated fluids. The inspectors determined that no contaminated spills, leaks, or unplanned releases had occurred since the last inspection in this area.

Analyses are performed for tritium and, for selected samples, hard-to-detect radionuclides for both groundwater and drinking water samples. No levels exceeding the EPA drinking water limit of 20,000 picocuries per liter (corresponding to 4 millirem per year to a member of the public) have been identified in the onsite or offsite environs.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization: The inspectors reviewed the plant's solid radioactive waste system description in the UFSAR and process control program (PCP). The most recent radiological effluent release report was reviewed for information on the types and amounts of waste disposed. The scope of the licensee's audit program was reviewed to verify that it met the requirements of 10 CFR 20.1101(c). The inspectors walked down the accessible portions of the liquid and solid radioactive waste processing systems to verify and assess that the current system configuration and operation agreed with the UFSAR and PCP. The use of video cameras to monitor batch tank levels was also discussed with selected radwaste and operations representatives.

The inspectors reviewed the radiological operating report for any documented changes to the radwaste processing systems and discussed observations from a walkdown of the systems with selected radwaste and operations representatives.

The inspectors reviewed the plant's process for transferring radioactive resin and sludge discharges into shipping/disposal containers to determine if appropriate waste stream mixing and/or sampling procedures and methodology for waste concentration averaging provided representative samples of the waste product for waste classification purposes. The inspectors reviewed current 10 CFR 61 analysis results and the procedures for obtaining the samples to support the analysis. The scaling factors used for radioactive waste streams and calculations used for determining the amount of hard to detect nuclides were reviewed. The program was reviewed to verify compliance with 10 CFR 61.55-56 and Appendix G of 10 CFR 20.

The inspector reviewed the program for provisions that would ensure that the waste stream composition accounted for changes in operational parameters and would remain valid between required periodic updates.

Transportation: The inspectors observed the preparation and shipment of resin to a vendor facility. The observations included packaging, surveying, labeling, placarding, vehicle checks, driver's briefing and emergency instructions, a review of shipping papers provided to the driver, and licensee final verification of shipment readiness. The inspectors observed two LSA-II type shipments, one involved a cask of dewatered condensate resin and the other was for contaminated plant protective clothing. The inspectors reviewed selected shipping documentation for shipments that had occurred from January 2008 to May 2009. The inspectors reviewed the QA surveillance documentation verifying compliance with the Certificate of Compliance for a Type B package that included dewatered resin. The inspectors observed, interviewed and reviewed the training records of the radwaste workers who were involved in the selected shipments.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution: Select PERs and self-assessments were reviewed in detail and discussed with cognizant licensee personnel. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1, Corrective Action Program, Rev. 15. Documents reviewed for problem identification and resolution are listed in Section 2PS2 of the report Attachment.

The inspectors completed the six required line-item samples described in IP 71122.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Mitigating Systems

Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the Performance Indicators (PI) listed below, including procedure SPP-3.4, Performance Indicator for NRC Reactor Oversight Process for Compiling and Reporting PIs to the NRC. The inspectors reviewed the raw data for the PIs listed below for the second quarter of 2008 through first quarter 2009 and discussed the methods for compiling and reporting the PIs with cognizant licensing, engineering, and maintenance rule personnel. The inspectors compared the licensee's raw data against graphical representations and specific values reported to the NRC in the second quarter 2009 PI report to verify that the data was correctly reflected in the report.

The inspectors also reviewed the past history of PERs for any that might be relevant to problems with the PI program. The inspectors reviewed Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to verify that industry reporting guidelines were applied.

- Unit 1 Safety System Functional Failures
- Unit 2 Safety System Functional Failures
- Unit 3 Safety System Functional Failures

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, 2008, through March 31, 2009, for the Occupational Exposure Control Effectiveness PI. For the reviewed period, the inspectors assessed CAP records to determine whether HRA, 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 reviewed documents relative to these PI reviews are listed in Sections 2OS1, 2OS2, and 4OA1 of the report Attachment.

b. Findings

No findings of significance were identified.

.3 Cornerstone: Public Radiation Safety

Radiological Control Effluent Release Occurrences

a. Inspection Scope

The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the period of April 1, 2008 through March 31, 2009. For the assessment period, the inspectors reviewed cumulative doses to the public, gaseous and liquid effluent release permits, and selected PERs related to effluent control. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. The reviewed data were assessed against guidance contained in NEI 99-02. Documents reviewed are listed in Section 4OA1 of the report Attachment.

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 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP implementation and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review included the results from daily screening of individual PERs (see Section 4OA2.1 above), licensee trend reports and trending efforts, and independent searches of the PER database and WO history. The review also included issues documented outside the normal CAP in system health reports, corrective maintenance WOs, component status reports, site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the six-month period of January 2009 through June 2009, although some PER database and WO searches expanded beyond these dates. Furthermore, the inspectors verified that adverse or negative trends identified in the licensee's PERs, periodic reports and trending efforts were entered into the CAP. Inspectors also interviewed the appropriate licensee management.

b. Findings and Observations

Inspectors reviewed the licensee's integrated trend review (ITR) program and the implementation of the program. As documented in Performance Improvement Department Procedure (PIDP) PIDP-11, PER Trending, trend reviews were to be conducted each quarter. The intent of this review was to identify the top organizational issues, both at the department and site level, and to report on the progress being made to resolve them. The inspectors determined that the licensee's trending program, in general, has shown improvement from previous inspections. The Integrated Trend Review Meeting was scheduled well in advance and departments were actively participating in the process. It should also be noted that a new procedure, PDIP-12, Integrated Trend Review, has been issued during this period, to provide additional guidance on conducting trending.

The inspectors noted areas in the trending program that warranted additional attention by the licensee, as follows:

- Some departmental negative trends that were continuing from previous ITRs relied on existing PERs or were closed to existing programs, without conducting a thorough evaluation if the existing PERs or programs were adequate to correct the problem. For example, several negative trends identified in the radiation protection department (e.g., contaminated area, number of hot spots, number of catch containments), were identified and new PERs were written. However, these PERs were closed to existing programs which have not been effective in reversing these trends. It was not apparent that an adequate evaluation was being conducted to ensure that the existing PER corrective actions were being effective, that these actions were timely enough or that criteria was being established where additional actions were needed (i.e., when an additional PER would be required).

- Department trend reports contained issues identified by external organizations, but in general did not address why the line organization did not identify these issues as required by PIDP-11. The licensee initiated PER 171857.

The inspectors conducted an independent review to identify potential negative trends, and identified several notable trends, as follows:

- A potential adverse trend was identified in conducting adequate PMTs. Some examples included: Unit 2 HPCI steam admission valve was repaired to correct seat leakage, but no seat leakage PMT was conducted (PER 161154); RHR HX outlet valves exhibited high vibrations causing repetitive failures, but following replacement of these valves vibration levels are not measured or evaluated (PER 171914); EECW manual isolation valve was replaced, but adequate flow to the room cooler was not verified during the PMT (PER 170650); and pressure testing following replacement of a manual isolation valve in the control rod drive system was not conducting at normal operating pressure (PER 172233). The licensee initiated PER 173055 to address this trend.

- The licensee established a PER corrective action closure review committee to ensure that corrective actions were being completed as described in the original action statement and properly documented. Over 70 PERs have been generated by this committee to document problems with PER action closures. In general, these PERs corrected the specific condition described, but no trend PER has been generated to evaluate and correct this trend in inappropriate approval of inadequate PER actions. The licensee initiated PER 175822 to address this trend.

- There have been several instances that inspectors have identified direct physical contact of safety related piping and/or conduits on other systems, structures or components. Examples included 1A Core Spray pump motor heater conduit against the 1A CS pump casing vent, 1C RHR pump seal leakage pipe against the 1C RHR pump suction vent line, 1A RHR pump seal water vent valve against 1A RHR pump seal leakage piping, and 1B CS cyclone separator tubing against the 1B CS pump pedestal. The licensee initiated PER 173522 to address this trend.

No violations of NRC requirements were identified.

.3 <u>Focused Annual Sample Review – Risk Management</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the specific corrective actions associated with the licensee's risk management trending PERs 160566, 167965, 170977, and other related PERs. Additionally, the inspectors reviewed corrective actions that resulted in issuing the new procedure SPP-7.3, Work Activity Risk Management Process; and a revision to existing procedure SPP-7.1, On-Line Work Management. Inspectors also reviewed BP-336, Risk Determination and Risk Management. Furthermore, the inspectors interviewed daily scheduling, outage, and operations risk management personnel.

b. Findings and Observations

Introduction: The inspectors identified an issue associated with the possible omission of certain risk significant systems from the licensee's routine assessment of online risk that was required by 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This issue is being characterized as an unresolved item (URI).

Description: Several systems identified as risk significant by the licensee's MR Program, and listed in Table 2, Initiators and System Dependency, of the "Risk Informed Inspection Notebook for Browns Ferry Nuclear Plant Units 1 and 2", did not appear to be considered as part of the licensee's routine online risk assessment. The inspectors observed that the following risk significant systems identified in Table 2 did not appear to be evaluated for on-line risk impact by the licensee: Raw Cooling Water (RCW), Fuel Oil Transfer, Emergency Diesel Generator (EDG) Room Ventilation (i.e., exhaust fans), Plant Control Air, Drywell Control Air, and Containment Ventilation. Furthermore, examples of risk significant systems identified as risk significant by the licensee's MR Program that did not appear to be assessed for online risk were as follows: RCW, EDG Room Ventilation, EDG Starting Air, and the Fuel Oil Transfer system.

To address the inspectors' concern regarding the assessment of online risk for all risk significant systems the licensee initiated PER 169904. As part of the corrective actions for PER 169904, the licensee committed to review Table 2 of risk significant systems against the list of risk significant systems as determined by the new Browns Ferry PRA model that was being developed, once it has been peer reviewed and approved. At which point the licensee's maintenance rule and on-line risk management tools would be updated accordingly.

This issue of concern requires additional information from the licensee. Such as, whether the MR and Table 2 designations of risk significant systems are appropriate, and how the current online risk assessment tools do or do not scope or bound the impact on risk associated with the unavailability of the aforementioned systems. Consequently, pending additional information from the licensee and further review by the NRC, this issue will be identified as URI 05000259, 260, 296/2009003-04, Inadequate Scoping of Risk Significant Systems for Online Risk Assessment.

Observations

The inspectors had the following observations:

PERs 160566, 167965, and 170977

The evaluation, development, and implementation of corrective actions for inadequate communications of plant risk were untimely and remained incomplete. As a potential adverse trend, this issue was discussed with the licensee in December 2008 and documented in NRC Integrated Inspection Report 05000259/2008005, 05000260/2008005 and 05000296/2008005, issued in January 2009. The licensee initiated PER 160566 on January 8, 2009 to document this potential trend in plant risk management where there were numerous instances in which risk assessments from ORAM, the shutdown plant outage risk assessment management software, and SENTINEL, the operating plant computer based risk management model, did not accurately reflect the actual plant equipment configuration. The corrective action to evaluate the potential trend was initiated on February 6, 2009. Subsequently, the licensee extended the PER citing resource re-allocation due to unplanned forced outages. On April 7, 2009, the licensee determined that a trend existed for not meeting expectations for clear communications of plant risk such as: conflicting risk levels on different site documents; incorrect information related to equipment out of service in the plan of the day; and inadequate transition from ORAM to SENTINEL risk model during unit restart. On that same day, PER 167965 was initiated to develop a corrective action plan for this trend. On May 6, 2009, PER 167965 was prematurely closed with incomplete actions taken, contrary to actions previously discussed with NRC inspectors. On May 11, 2009, the licensee initiated a third PER 170977 to document additional corrective actions. Inspectors discussed their concern with further prolonging corrective actions and the licensee initiated PER 171722 for inadequate and untimely corrective actions.

<u>SPP-7.3, Work Activity Risk Management Process</u>

Inspectors reviewed newly developed licensee procedure SPP-7.3, Work Activity Risk Management Process, to evaluate the corrective actions documented in PER 167965. The inspectors made the following observations:

- The relocation of risk management action (RMA) guidance from SPP-7.1, On-Line Work Management, to SPP-7.3 was incomplete. SPP-7.3, Appendix E lists RMAs available for various levels of risk, but did not include all of the RMAs in SPP-7.1, Section 3.5.1.c. Additionally, SPP-7.1, Appendix J contained critical evolution meeting guidance that in itself was an RMA, and was not contained in SPP-7.3. The licensee’s site and corporate (SPP administrators) personnel agreed with inspector observations and intend to revise the two SPPs in accordance with their corporate procedure process.

- Risk management review forms to support the new SPP-7.3 procedure had still not been developed four weeks after the procedure became effective. The licensee stated that implementation of the procedure was planned to be a phased-in effort coinciding with the work week development process, and initiated an action as part of PER 170977 to track the development of the forms.

<u>BP-336, Risk Determination and Risk Management</u>

With approximately 11 weeks remaining until BP-336 would be phased out and deactivated, inspectors noted that Operations guidance associated with plant risk matrices and protected equipment (which the operations department intended to maintain) did not have a prospective home in another procedure or similar guidance document. Inspectors also noted that the licensee had not documented this concern in their CAP, and therefore a documented plan of action was not in place. The licensee initiated an action as part of PER 170977.

Equipment Out of Service (EOOS) Risk Assessment Tool

During discussions with the licensee concerning their development of new risk management and assessment tools, inspectors noted that with the planned phasing in of the new PRA-based EOOS software near the end of 2009 and the planned phasing out of SENTINEL, there was a lack of risk assessment tool coverage for Mode 3 Hot Shutdown. The licensee agreed that this would be the case until the low power shutdown model was completed in approximately five years which would apply in Modes 1, 2, and 3. The licensee initiated PER 169925 to resolve risk tool coverage for Mode 3 conditions.

4OA3 Event Follow-up

.1 (Closed) LER 05000260/2009-001, Manual Reactor Scram Following Stator Cooling Water Equipment Failure

a. Inspection Scope

On February 16, 2009, the Unit 2 reactor was manually scrammed from 100 percent power due to high main generator stator cooling water (SCW) temperature. The loss of stator water cooling was due to the failure of the SCW temperature controller which caused the temperature control valve (2-TCV-035-0054) to fail open and bypass too much SCW from the SCW heat exchanger. The exact cause of the SCW temperature controller failure was not definitively determined by the licensee. During and following the scram, all safety-related mitigating systems operated as designed, and all operator actions in response to the scram were deemed to be appropriate (see IR 05000260/2009-002, Section 4OA3.1). This LER and its associated PER 163680, including the root cause analysis (RCA), were reviewed by the inspectors.

b. Findings

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

.2 (Closed) LER 05000259/2009-001, Turbine Trip and Reactor Scram Due to Power Load Unbalance Signal on the Main Generator

a. Inspection Scope

On February 18, 2009, the Unit 1 reactor was automatically scrammed from 100 percent power due to a power load unbalance when the main turbine generator (MTG) output breaker tripped. The Unit 1 MTG protective relays were actuated when a significant quantity of water was blown onto the main generator output isophase buses causing an electrical ground fault. The source of the water was from excessive condensation that had been accumulating in the idle 1B isophase bus duct cooling fan, such that when operators switched from the in-service 1A fan, to the 1B fan, a large quantity of water was entrained and ejected onto the isophase buses. During and following the scram, all safety-related mitigating systems operated as designed, and all operator actions in response to the scram were deemed to be appropriate (see IR 05000259/2009-002, Section 4OA3.2). This LER and its associated PER 163815, including the RCA, were reviewed by the inspectors. The inspectors also reviewed previous PERs related to excessive condensation and walked down the Unit 1 isophase bus cooling (IBC) system shortly after the event. Furthermore, the inspectors interviewed the RCA team leader.

b. Findings

This LER is considered closed with one finding identified.

Introduction: A Green self-revealing finding was identified for a failure to implement corrective actions in a timely manner to address excessive IBC system condensation that resulted in an MTG trip and Unit 1 reactor scram caused by water accumulation in the IBC ductwork, which created an electrical ground fault on the MTG isophase busses.

Description: A review of work orders from as far back as 1993, by the licensee, determined that plant personnel frequently observed excessive condensation dripping from the IBC system ductwork during winter operating conditions. Alabama winters have frequently included certain days when river water temperatures were very cold, but ambient air was warm and humid (e.g., after a storm front). During these conditions, the IBC coolers would generate considerable quantities of condensed water in the bus ductwork. Excessive condensation dripping from the IBC ductwork seams had been observed and tolerated by the licensee for many years.

In 2006, Unit 1 installed a new IBC system. This new system was designed to have much a higher capacity cooler, along with dual fans vice the original single fan design. The new Unit 1 IBC design was based on the worst case Extended Power Uprate Isophase Bus loading, along with worst case river water temperatures (i.e., during the summer). The licensee determined in their root cause analysis, that the Unit 1 IBC design change did not consider or evaluate the potential adverse implications associated with winter operating conditions. The inspectors verified that the technical considerations required to be evaluated by SPP-9.3, Plant Modifications and Engineering Change Control, for design changes did not require the impact of seasonal environmental changes to be evaluated. However, it was apparent to the licensee and inspectors, that the new design features of the Unit 1 IBC system were more conducive to generating and accumulating larger quantities of condensation during winter operating conditions than the old system. On December 8, 2007, as in previous occasions, plant personnel observed water leaking from the seams of the Unit 1 IBC ductwork and initiated WO 07-727187-000 to correct the problem. However, the WO wasn't worked until October 2008 at which time there was no water to be found. This WO was closed with no work performed. Then, on December 10, 2008, plant personnel again discovered excessive moisture coming out of the seams of the Unit 1 IBC ductwork. This time the abnormal equipment condition was entered into the licensee's CAP as PER 158940 (Level C PER). The inspectors noted that the specific actions to promptly correct the problem recommended by the PER were not implemented prior to the event.

In the purpose section of SPP-3.1, Corrective Action Program, it stated that this SPP established the processes and responsibilities for documenting and resolving problems at the station. This program procedure also addressed safety-related problems required to be corrected by 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. The overall purpose of the licensee's CAP was to provide the necessary site standards to ensure that all abnormal equipment conditions entered into the CAP, including non-safety related equipment deficiencies, were resolved. The licensee initiated PER 158940, and designated it as a Level C PER, to ensure that the site standards imposed by SPP-3.1 were implemented and followed to resolve the excessive IBC system condensation. The licensee's failure to implement the actions recommended by PER 158940 in a timely manner was a missed opportunity to prevent the Unit 1 reactor scram of February 18, 2009. During the subsequent event investigation, shortly after the Unit 1 scram, the licensee discovered approximately 35 gallons of water (i.e., condensate) remaining in the IBC ductwork. From this observation and a system walkdown, the licensee concluded that at least this amount of water had originally condensed and collected in the idle 1B IBC fan housing prior to the event. Based on the observed persistent problems with water accumulation in the IBC ductwork, including the multiple WO's and PERs that were written, the inspectors concluded the licensee had many past and recent opportunities to address this condition. It was therefore reasonable to conclude that it was within the licensee's ability to foresee and correct the condition before it resulted in a reactor scram.

Analysis: The licensee's failure to implement corrective actions in a timely manner, according to the site standard of SPP-3.1, in order to resolve a known abnormal equipment condition involving excessive IBC condensation, was a performance deficiency which directly resulted in an automatic reactor scram. This finding was determined to be greater than minor because it was associated with the initiating events cornerstone attribute of equipment performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available.

The cause of this finding was directly related to the cross-cutting aspect of appropriate and timely corrective actions in the area of Problem Identification and Resolution, because the license had identified an abnormal equipment condition related to excessive IBC system condensation for which corrective actions were specified but not carried out (P.1.d).

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of a regulatory requirement since the IBC system is not safety-related. Because this finding does not involve a violation of regulatory requirements, was entered in the licensee's CAP as PER 163815, and has very low safety significance, it is identified as FIN 05000259/2009003-05, Untimely Actions to Resolve Excessive IBC System Condensation Results in Unit 1 Reactor Scram.

.3 (Closed) LER 05000259/2009002-00, Unexpected Logic Lockout of the Loop II Residual Heat Removal System Pumps

a. Inspection Scope

On March 21, 2009, RHR pumps 1B and 1D automatic start feature was locked out during a surveillance, while RHR pumps 1A and 1C had previously been removed from service for testing. The cause of the lockout was an error in the Core Spray Logic Functional Test procedure that resulted in the installation of a jumper in the wrong logic panel. Operations personnel immediately recognized that the placement of the jumper had rendered the pumps inoperable and entered TS LCO 3.0.3. After approximately one minute, the automatic start function of RHR Pumps 1B and 1D was restored and TS 3.0.3 was exited. Corrective actions included removal of the jumper, revision of the surveillance procedure and placing similar logic functional test procedures on administrative hold until a thorough review could be conducted. This event was reviewed by the inspectors and documented in Inspection Report 05000259/2009-002. NCV 05000259/2009002-02, Inadequate Surveillance Procedure Causes Loss of Unit 1 RHR System Safety Function, was issued as a result of this review. The inspectors reviewed the LER and associated PER 166487 and root cause analysis.

b. Findings

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

4OA6 Meetings, Including Exit

.1 Quarterly Integrated Inspection Report Exit Meeting Summary

On July 6 and 17, 2009, the senior resident inspector presented the inspection results to Mr. Jim Randich, Mr. Don Grissette, and other members of the licensee staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation.

- TS 5.7.1, requires that each individual or group entering such an area (locked high radiation area) shall possess a device that continuously transmits dose rate information to a remote receiver monitored by radiation protection personnel or be under the surveillance of an individual qualified in radiation protection procedures equipped with a radiation monitoring device that continuously displays dose rates in the area. Contrary to this, on April 15, 2008, a radworker retrieved his ED from the 550 foot elevation of the U3 drywell without being issued another ED, being monitored by a RPT or being accompanied by a RPT with a survey instrument. This event was documented in the licensee's CAP as PER 142319. This finding was of very low safety significance because there was no substantial potential for overexposure and the licensee's ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. Baker, Operations Unit Supervisor
S. Berry, Systems Engineering Manager
S. Bono, Director of Engineering
M. Button, Maintenance Manager
S. Cephus, Component Engineering Supervisor
P. Chadwell, Operations Manager
J. Emens, Site Licensing Supervisor
D. Feldman, Operations Support Superintendent
A. Feltman, Emergency Preparedness Manager
E. Frevold, Design Engineering Manager
R. Glisson, Maintenance Support Manager
F. Godwin, Licensing Manager
D. Grissette, Director of Site Technical Support
L. Hughes, Operations Superintendent
J. McCarthy, Director of Safety and Licensing
M. McLaughlin, Unit Supervisor
J. Mitchell, Site Security Manager
M. Palmer, Assistant Plant Manager
R. Perry, Fire Protection Shift Supervisor
B. Pierce, Chemistry/Environmental Manager
B. Quinn, Daily Scheduling Manager
E. Quinn, Performance Improvement Manager
J. Randich, General Manager of Operations
R. Rogers, Modifications and Projects Manager
P. Sawyer, Radiation Protection Manager
J. Underwood, Site Nuclear Assurance Manager
J. Walton, Radiation Protection Supervisor
R. West, Site Vice President

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000259, 260, 296/2009003-04	URI	Inadequate Scoping of Risk Significant Systems For Online Risk Assessment (Section 4OA2.3)

Opened and Closed

05000260/2009003-01	NCV	Failure to Assess and Manage Shutdown Risk Associated with Outage Maintenance Activities (Section 1R13)
05000259/2009003-02	NCV	Failure to Perform an Adequate Risk Assessment during Severe Weather Conditions (Section 1R13)
05000259, 260, 296/2009003-03	NCV	Failure to Comply with the Requirements of an RWP by Entering a Posted High Radiation Area (Section 2OS1)
05000259/2009003-05	FIN	Untimely Actions to Resolve Excessive IBC System Condensation Results in Unit 1 Reactor Scram (Section 4OA3.2)

Closed

LER 05000260/2009001-00	LER	Manual Reactor Scram Following Stator Cooling Water Equipment Failure (Section 4OA3.1)
LER 05000259/2009001-00	LER	Turbine Trip and Reactor Scram Due to Power Load Unbalance Signal on the Main Generator (Section 4OA3.2)
LER 05000259/2009002-00	LER	Unexpected Logic Lockout of the Loop II Residual Heat Removal System Pumps (Section 4OA3.3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

0-GOI-200-3, Hot Weather Operations, rev. 9
IGA-6, Power System Operations, Rev. 11
SPP-7.1, On-Line Work management, Rev. 13
0-GOI-300-4, Switchyard Manual, Rev. 74
NOM-SDP-7-SYR-Work Plan, Work Plan and Development and Performance for Switchyard Risk Activities, Rev. 4
TRO-VP-SPP-10.006, Loss of SCADA and/or EMS, Rev. 3
TRO-TO-SOP-10.128, Browns Ferry Nuclear Plant (BFN) Grid Operating Guide
TRO-TO-SOP-10.328, Nuclear Offsite Disqualification Notification and Call-Out Procedure, Rev. 5

Section 1R04: Equipment Alignment

2-OI-75 Attachment 1, Core Spray System Valve Lineup Checklist, March 19, 2007
2-OI-75 Attachment 2, Core Spray System Panel Lineup Checklist, March 19, 2007
2-OI-75 Attachment 3, Core Spray System Electrical Lineup Checklist, March 19, 2007
Drawing 3-47E812-1, Flow Diagram High Pressure Coolant Injection System
3-OI-74, Residual Heat Removal System, Rev. 86
3-OI-74, Attachment 1, RHR System Valve Lineup Checklist, Rev. 86
3-OI-74, Attachment 2, RHR System Panel Lineup Checklist, Rev. 86
3-OI-74, Attachment 3, RHR System Electrical Lineup Checklist, Rev. 86
Drawing 3-47E811-1, Flow Diagram Residual Heat Removal System, Rev. 64

Section 1R05: Fire Protection

Fire Protection Report, Volume 1, Section 2 Fire Hazards Analysis, Rev. 4
Fire Protection Report, Volume 2, Section IV.3, Pre-Plan No. RX1-621, Rev. 8
Fire Protection Report, Volume 2, Section IV.6, Pre-Plan No. RX2-639, Rev. 8
Fire Protection Report, Volume 2, Section IV.9, Pre-Plan No. RX3-593, Rev. 8
Fire Protection Report, Volume 2, Section IV.9, Pre-Plan No. RX3-621, Rev. 8
Fire Protection Report, Volume 2, Section IV.9, Pre-Plan No. RX3-639, Rev. 8
Fire Protection Report, Volume 2, Section IV.11, Pre-Plan No. CB2-593, Rev. 8
Fire Protection Impairment 08-1840, Door 651
Fire Protection Impairment 09-1898, Scaffold
Fire Protection Impairment 09-1920, Hourly Roving Watch SSI Manual Actions
Fire Protection Impairment 09-1935, B3 EECW Pump Tagged Out
Fire Protection Impairment 09-1819, Unit 2 Outage Scaffolding
Fire Protection Impairment 09-1920, Hourly Roving Watch SSI Manual Actions
Fire Protection Impairment 09-1939, A3 EECW Pump Tagged Out
Fire Protection Impairment 09-1935, B3 EECW Pump Tagged Out
0-SI-4.11.G.1.b(4), Visual Inspection of Fourth Period Appendix R Fire Dampers, Rev. 11
0-SI-4.11.G.1.b(5), Visual Inspection of Fifth Period Appendix R Fire Dampers, Rev. 5
Drawing 3-47E3392-627, Fire Protection 10CFR50 Appendix R Penetration Seal Tabular Drawing EL 621.25 Details J, L, M, N, P & Q, Rev. 4

Drawing 0-47W391-2, Fire Protection 10CFR50 Appendix R Penetration Seal Typical Details Flexible Fire Seals FB-1, -1A, -2, & -3, Rev. 0

Section 1R08: Inservice Inspection

Corrective Action Documents
PER 163525, EPU Steam Dryer Analysis – T Beam, 02/12/2009
PER 166823, Engineering Function Evaluation of the Reactor Cavity Head for Unit 2, 03/25/2009
PER 147298, 2-FCV-64-31 Closure Stroke Time, 06/25/2008
PER 152782, Linear Indications, 09/18/2008
PER 141734, 2A RHRSW Inlet Header Pinhole Leak, 04/07/2008
PER 146170, Steam Leak on 2-FCV-001-00146, 06/01/2008
PER 121787, Indication in N9 Nozzle to Vessel Weld, 03/17/2007
PER 121003, NOI U2C14-017, CRD Weld Indication, 03/06/2007
PER 169585, RHRSW Effluent, 04/26/2009

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
2-SI-4.6.G Rev 36, Inservice Inspection and Risk-Informed Inservice Inspection Program Unit 2, 04/10/2009
ISI Report# R117, RPV Nozzle Ultrasonic Examination Summary Sheet
Report# R074, Examination Summary and Resolution Data Sheet
MDQ206820070013 Rev 0, N-9 Nozzle Weld Flaw Evaluation per IWB-3600
WCAP-16845-NP Rev 0, Metallurgical Investigation of Recorded Indications at the Check Valve to Pipe Dissimilar Metal Weld at Browns Ferry Unit 2 Nuclear Generating Station
Digital Thermometer Calibration Sheet, Calibration No. 522350, 08/16/2008
Digital Thermometer Calibration Sheet, Calibration No. 558272, 06/20/2008
Light Meter Calibration Sheet, Calibration No. E31629, 06/18/2008
Report No. R041, Record of Liquid Penetrant Exam
Report No. R049, Examination Summary and Resolution Sheet
Certified Test Report, Spotcheck Developer SKD-S2, Batch 05A10K, 02/01/2005

Certified Test Report, Spotcheck Cleaner/Remover SKC-S, Batch 07A06K, 01/30/2007
Certified Test Report, Spotcheck Penetrant SKL-SP1, Batch 05G02K, 07/15/2005
Certified Test Report, Ultragel II Batch 06125, 04/18/2006
Certification of Test, Calibration Block WB-78, 03/27/2002
Certificate of Calibration, Phasor XS Flaw Detector, Serial No. E41820, 08/06/2008
Certification of Conformity, Probe 01XC90, 03/18/2009
WO# 09-711929-000, Replace Valve BFR-1-SHV-067-0655
WPQR GT11-SPEC-1, Procedure Qualification Record, 12/29/1987
WPQR GT11-0-1A, Procedure Qualification Record, 01/26/1981
WPQR GTA18-B-1, Procedure Qualification Record, 03/01/2004

Section 1R11: Licensed Operator Requalification Program

OPDP-1, Conduct of Operations, Rev. 9
OPL173S280, Simulator Evaluation Guide, Group 6 Isolation and Main Steam Line Break in Containment, Rev. 1
2-AOI-64-2D, Group 6 Ventilation System Isolation, Rev. 30
2-EOI-1 Flowchart, RPV Control, Rev. 12
2-EOI-2 Flowchart, Primary Containment Control, Rev. 10

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
PER 160047, Unplanned LCO Entry due to Ground on HPCI Logic
PER 160537, HPCI Declared Inoperable Following Testing of 2-FCV-73-16
PER 152914, Maintenance Rule (a)(1) Status for Unit 2 HPCI
PER 144253, Unit 2 HPCI Oil Sample High Moisture
PER 137060, Unit 2 HPCI Steam Admission Valve Wedge Binding
PER 137859, U2 HPCI Steam Admission Valve Pressure Seal Ring Failure
PER 137687, Wrong Parts used for U2 HPCI Steam Admission Valve Repair
Unit 2 Function 073-B (a)(1) Plan, Rev. 0
CDE 663, 2-FCV-073-0016 Extended Maintenance for Seat Leakage
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MREP Meeting Minutes dated 3/13/2008
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PER 162961, Unit 1 RCIC in Maintenance Rule (a)(1) Status
PER 158304, U1 RCIC Governor Valve Failed to Re-open Following Flow Test
PER 158928, 1-FCV-71-3, RCIC Outboard Steam Supply Valve Failed to Open During Testing
Unit 1 Function 071-B (a)(1) Plan, Rev. 0
CDE 729, 1-FCV-071-0010 Binding
CDE 730, 1-FCV-71-3 Fails to Open
MREP Meeting Minutes dated 3/19/2009
MPI-0-071-TRB001, Reactor Core Isolation Cooling (RCIC) Turbine Preventative Maintenance, Rev. 24

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0-TI-367, BFN Equipment to Plant Risk Matrix, Rev. 10
BP-336, Risk Determination and Risk Management, Rev. 7
SPP-7.1, On Line Work Management, Rev. 12
SPP-7.2, Outage Management, Rev. 13
SPP-7.3, Work Activity Risk Management Process, Rev. 0

PER 168316, Sentinel Program Not Posted
Browns Ferry Plan of the Day Dated April 9, 2009
Sentinel Runs Dated April 6-9, 2009
Browns Ferry Plant 12 Week Rolling Schedule T-0 Summary for WW 2915

BP-336 Plant Protected Equipment report (4/21 and 4/22/09)
Unit 2 Sentinel reports (4/21 and 4/22/09)
T-0 Summary Report for Work Week #2917
PER 169448, Contingency Actions Not Performed in a Timely Manner for OOS Risk Sig Equip

BP-336 Plant Protected Equipment report (5/4/09)
Unit 2 Sentinel reports (5/4/09)
Unit 1 Daily Plant Status Report (5/4/09)
T-0 Summary Report for Work Week #2919
PRA Evaluation Response – BFN-0-09-029

PER 170184, Installation of N-1 Nozzle
PER 170184, Diesel Removed From Service During Severe Weather

ORAM Reports (5/19 and 5/20/09)
Unit 2 Outage Reports (5/19 and 5/20/09)
BP-336 Protected Equipment Log
ORAM Logic Database Report for Shutdown Cooling
Unit 2 Decay Heat Removal Decay Curves
PER 171728, Unplanned Entry into ORAM Orange Condition due to B Secondary ADHR Pump Failure

Section 1R15: Operability Evaluations

PER 169397, Flow Error in FE 43292
PER 167344, EECW Low Flow to 2B Core Spray Room Cooler
0-TI-54, EECW System Operational Flush, Rev. 9
2-SI-3.2.4, EECW Check Valve Test, Rev. 39
General Design Criteria BFN-50-7075, Core Spray Cooling System, Rev. 11
General Design Criteria BFN-50-7067, Emergency Equipment Cooling Water System, Rev. 17
General Design Criteria BFN-50-7064B, Reactor Building Ventilation System, Rev. 11
Calculation MD-Q0067-930043, RHR and Core Spray Room Cooler Analysis, Rev. 6
Calculation MD-Q0067-2005-0027, Flow Element Constants for EECW Flow Elements, Rev. 0
Drawing 2-47E859-1, Emergency Equipment Cooling Water System Flow Diagram, Rev. 29
FSAR Section 10.10, Emergency Equipment Cooling Water System, BFN 22
Technical Specifications and Bases 3.7.2, Emergency Equipment Cooling Water and Ultimate Heat Sink, Amendment 255

PER 163879, 0-SR-3.8.1.8(I) Acceptance Criteria Step Not Satisfied

PER 164013, Unit 1 Main Turbine TGOP Drawing and Wiring Discrepancies
PER 164083, 480V Load Shed Reportability Determination
PER 166870, FE on TGOP Past Operability
Drawing 0-45E602-5, Wiring Diagram Turbo-Generator Auxiliaries Schematic Diagram, Rev. 8
Drawing 1-0124B3723, Main Turbine Gear Oil Pump Sub No. 20, Rev. 0
Drawing 1-45E1748-4, Wiring Diagram 480V Shutdown Board 1B Connection Diagram, Rev. 9
WO 09-711952-000, Troubleshoot U1 TGOP Failure to Load Shed
0-SR-3.8.1.8(I), 480V Load Shedding Logic System Functional Test (Division I), Rev. 11
Calculation EDQ005720020069, Diesel Load Study for Units 1 and 2, Rev. 13
Calculation EDQ0057920034, 4.16KV and 480V Bus Load, Voltage Drop and Short Circuit Calculation, Rev. 42
Technical Specification and Bases 3.8.1, Electrical Power Systems, AC Sources – Operating, Units 1 and 2, Amendment 249 and Revision 52 respectively
FSAR, Section 8.5 Standby AC Power Supply and Distribution, BFN 22

PER 147819, Steam Leak in 3B SJAE Room
PER 147858, Implementation of Safety Procedure 806 for 3B SJAE Room Leak
PER 147854, Potential Water Entry into 3A CBP Oil System
Drawing 3-47E801-2, Flow Diagram Main Steam, Rev. 27
Drawing 3-47W405-8, Mechanical Offgas System, Rev. 6
Drawing 0-47W400-8, Mechanical Main Steam Piping, Rev. 1
Drawing 0-47W400-5, Mechanical Main Steam Piping, Rev. 3
Drawing 0-47W400-4, Mechanical Main Steam Piping, Rev. 5
WO 08-718477-001, 3B SJAE Leak Repair
3-SI-3.2.29, MSIV Alternate Leak Path Testing, Rev. 5
3-SI-3.2.30, MSIV Alternate Leak Path – Cold Shutdown Testing, Rev. 3
Calculation MD-Q0001-000006, Cracking Pressure for Check Valves 1/2/3-BFN-CKV-001-0742 and -744, Rev. 1
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NEDC-31858P-A, BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, August 1999

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Unit 2 Technical Specifications Section 3.3.3.2, Backup Control System
PER 169830, Compliance Instrument 1-LI-64-54B Out of Tolerance
PER 172826, Inaccurate Unit 1 Suppression Pool Level Indication at Backup Control Panel
Functional Evaluation 43376 (for PER 169830), Loops 1-LI-64-54 and 1-LI-64-66 Do Not Track
WO 09-714949-003, Perform Flush and Troubleshoot 1-LI-64-54 and 1-LI-64-66
WO 09-714949-004, Troubleshoot Level Readings of 1-LI-64-54 and 1-LI-64-66

PER 171208, Unacceptable Coatings Degradation in Unit 2 Primary Containment
Functional Evaluation 43508 (for PER 171208), Uncontrolled Coatings Exceed Design Limit
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MDQ099920060011, Transient NPSH/Containment Pressure Evaluation of RHR and CS Pumps
UFSAR Section 5.2, Primary Containment System
UFSAR Section 6.5.5, Potential Plugging of Emergency Core Cooling System Suction Strainers

Section 1R18: Plant Modifications

DCN 69528, Replace ASCO Solenoid Valves With Similar AVCO Valves, Rev. A
FSAR Section 5.3 Secondary Containment System, BFN-22
Technical Specifications and Bases 3.6.4.1 Secondary Containment, Amendment 290
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Technical Requirements Manual for Browns Ferry Nuclear Plant, Appendix A, Power Operated Secondary Containment Isolation Valves, Rev. 0
General Design Criteria Document 7064B, Reactor Building Ventilation System, Rev. 13
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General Design Criteria Document 7200B, 120 VAC Power Supply and Distribution, Rev. 11
General Design Criteria Document 7307, Post Accident Monitoring, Rev. 10
Drawing 3-47E865-12, Reactor Building Ventilation Flow Diagram, Rev. 45
PER 153130, Unit 2 Reactor Zone Fan Damper 2-FCO-64-43 Failed Open
WO 08-724013-018, Unit 3 Refuel Zone Exhaust Duct Outboard Isolation Damper ASCO Solenoid Replacement With AVCO
WO 08-724013-019, Unit 3 Refuel Zone Exhaust Duct Inboard Isolation Damper ASCO Solenoid Replacement With AVCO
0-SR-3.6.4.2.1, Secondary Containment Isolation Valve Stroke Timing, Rev. 12
EDQ2999920313, Mini-Calculation for Voltage Drop and Load Analysis for Modification on the 120v System of Unit 1 and 2, Rev. 98

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PER 173527, No Briefing Sheet Attached as Required
PER 173937, Temporary Monitoring Equipment in Noncompliance with MMDP-1 and SPP-9.5
PER 173939, 0-TI-230V Does Not Reference MMDP-1 or SPP-9.5
SPP-9.5, Temporary Alterations, Rev. 8
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WO 08-715494-000, Installation of Vibration Equipment on Unit 3 RHRSW HX Outlet Valves
0-OI-23, Residual Heat Removal Service Water System, Rev. 87
0-TI-230V, Vibration Monitoring, Rev. 6
0-TI-405, Plant Modifications and Design Change Control, Rev. 0
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0-SI-4.5.C.1(4), EECW Annual Flow Rate Test, Rev. 40
0-SI-3.1.11, EECW Pump Baseline Data Acquisition and Evaluation, Rev. 25
0-SI-3.1.4, EECW Pump Performance, Rev. 47
0-TI-345, EECW Pump Curve Data Acquisition, Rev. 5
0-TI-362, Inservice Testing of Pumps and Valves, Rev. 22

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
MCI-0-023-PMP002, EECW and RHRSW Pump (Byron Jackson Type KX) Disassembly, Inspection, Rework and Reassembly, Rev. 47
MCI-0-023-PMP003, EECW and RHRSW Pump (Byron Jackson Type KX) Removal and Reinstallation, Rev. 10
MCI-0-023-PMP004, EECW and RHRSW Pump Impeller Adjustment, Rev. 3
ECI-0-000-MOT001, Removal and Installation of AC and DC Motors, Rev. 54
EPI-0-000-TST001, Bridge, Megger and High Potential Testing of Electrical Equipment, Rev. 56
Work Order 08-716204-000, Replace Upper Column Shaft on A3 EECW Pump
Work Order 08-716204-001, MSB Support for removal of A3 EECW Pump Motor
Work Order 09-712134-000, Replace A3 EECW Pump with refurbished Pump
Work Order 08-712134-001, MSB Support for removal of A3 EECW Pump Motor
Work Order 09-714533-000, Adjust Impeller on A3 EECW Pump
Work Order 09-715933-000, Remove A3 EECW Pump for Inspection
Work Order 09-715933-001, MSB Support for removal of A3 EECW Pump Motor
Work Order 09-715933-003, Support for Testing of A3 EECW Pump
Evaluation 09-0-IST-023-389, Evaluation of ASME OM Code IST Test Results for A3 EECW Pump
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Drawing 1-47E858-1-ISI, ASME Section XI RHR Service Water System Code Class Boundaries, Rev. 24
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PER 35566, Tolerance for Setting RHR and CS Pump Flow Rates During Section XI Tests Cause Total Accuracy To Exceed 2percent
PER 166464, A3 EECW Pump Has Audible Knocking Sound
PER 167222, Unplanned Unavailability on the A3 EECW Pump
PER 167411, A3 EECW Pump Strainer Basket Damaged
PER 167758, Alignment Issues on A3 EECW Pump Due To Tight Packing
PER 168420, A3 EECW Pump Failed During 3-SI-4.5.C.1(2)
PER 168967, A3 EECW Pump Repeat Failure during PMT Surveillance
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PER 174373, Baseline Testing of A3 EECW Pump Could Not Be Used Due To Not Obtaining a Data Point at Design Flow Rate.
PER 174469, 0-SI-4.5.C.1(4), EECW System Annual Flowrate Not Initially Performed After Replacement of A3 EECW Pump on 4/7/2009
PER 174835, Procedure Revision Failed to Address Question of Operability of Non-Tested Pump
PER 174914, Confusion and Inconsistency in the Use of “N/A” During Surveillances
PER 174991, 0-TI-345 Steps Performed Out-of-sequence
PER 175117, A3 EECW Pump Testing Improvements
PER 175238, No 10CFR50.59 Screening Performed for Change to EECW Surveillance Procedure
PER 175244, Improvements in Quality Needed in Procedure 0-TI-345
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

PER 175255, Alert and Action Required Flows Reduced Due to Instrument Inaccuracy

WO 08-712349-000, Insert Riser Isolation Valve Packing Replacement
MCI-0-000-GTV001, General Maintenance Instructions for Gate Valves, Rev.21
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Drawing 2-47E820-2, Flow Diagram Control Rod Drive Hydraulic System, Rev.14
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WO 09-711929-000, North Header Supply to Core Spray II Cooler Valve Replacement
WO 08-724315-000, Monthly Monitoring of 1B CS Room Cooler Flow
0-OI-67, Emergency Equipment Cooling Water System, Rev. 85
PER 170170, North Header Supply to NE Core Spray Room Cooler Structural Integrity
PER 170453, Div II RHR Room Cooler Vent Valve Failure
PER 170454, A3 EECW Pump Operated Below Minimum Flow
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1-SI-3.2.4, EECW Check Valve Test, Rev. 31
Tagout 0-TO-2009-0001, Clearance 0-067-0010, 1-SHV-067-0655

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Work Order 07-721562-003, Uncouple and Couple 2D CS Pump from Motor
Work Order 07-721562-005, Perform PMTs Following Replacement of 2D CS Pump Motor
Work Order 07-721562-007, Perform Bridge and Megger on 2D CS Pump Motor
2-SR-3.5.1.6(CS II-Comp), Core Spray Loop II Comprehensive Pump Test, Rev. 3
ECI-0-000-MOT001, Removal and Installation of AC and DC Motors, Rev. 54
EPI-0-000-TST001, Bridge, Megger and High Potential Testing of Electrical Equipment, Rev. 55
MCI-0-075-PMP001, Core Spray Bingham Centrifugal 12x16x14 ½ One Stage Disassembly, Inspection, Rework and Reassembly, Rev. 8
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WOs 07-722378-008, -012, and -014, RHRSW HX Outlet Valve Replacement/Modification
WO 09-716814-000, Vibration Testing on RHRSW HX Outlet Valve 23-52
ECI-0-000-MOV009, Testing of Motor Operated Valves Using MOVATS Universal Diagnostic System (UDS) and Viper 20, Rev. 21
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2-OI-1, Main Steam System, Rev. 45
2-GOI-100-3A, Refueling Operations (RX Vessel Disassembly and Floodup), Rev. 37
2-GOI-100-3B, Refueling Operations (Reactor Cavity Letdown and Vessel Re-Assembly), Rev. 48
2-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor Vessel and Associated Piping, Rev. 25
2-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring, Rev. 18
2-SR-3.3.2.1.2, RWM Functional Test for Startup, Rev. 3
2-SR-3.3.2.1.7, RWM Program Verification, Rev. 2
2-SR-3.1.3.5(A), Control Rod Coupling Integrity Check, Rev. 20
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WO 08-724100-000, Electrical Maintenance PM for Monitoring CS Motor Bearing and Winding Temperatures
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RCI-9.1, Radiation Work Permits, Rev. 57
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RCI-31, HI-STORM Five Year Shielding Effectiveness Surveys, Rev. 0
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RCI-40.3, RP Actions for Operation's Unit 3 Procedural Hold Points, Rev. 8
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08380028, U3C13 DW Radcon Support (LHRA, Resp, Various Dress)
08382236, U3C13 DW Cut-out Replace 73-2 (LHRA Various Dress)
09270045, U2C15 Rx Bldg Scaffold Support (High Rad/Various Dress)
09270081, U2C15 Rx Bldg Movats (Various Dress)
09280025, U2C15 Drywell Radcon Outage Support (Dose Control/High Rad/Various Dress)
09280095, U2C15 DW ISI, IWE, Fac (High Rad/Various Dress)
09290018, U2C15 Refuel Floor Steam Dryer Tie Bar Repair (LHRA, Extremity & Multibadging)

Radiation Surveys
051209-32, Unit 2 Drywell 628’
051209-31, Unit 2 Drywell 616’
051209-29, Unit 2 Drywell 604’
051109-36, Unit 2 Drywell 584’
051209-20, Unit 2 Drywell 563’
042809-61, Unit 2 Drywell 550’
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041508-9, Unit 3 RXB 565’ General Area
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051808-25, Unit 3 TB 586’ 3B SJAE Room
052008-26, Unit 3 TB 586’ 3B SJAE Room
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050809-16, Low Level Module Warehouse
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032509-1, ISFSI Pad
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040709-19, ISFSI Pad
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PER 141446, Dose rate alarm
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PER 145551, 3B SJAE dose rate alarm
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Procedures, Manuals, and Guidance Documents
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RCI-26, Radiation Protection Department Standards and Expectations, Rev. 16
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SPP-7.0, Work Management, Rev. 1
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SPP-7.2, Outage Management, Rev. 12

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APR 08-0110, U1C7 Outage - Scaffolding Support, Rev.0
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APR 08-0124, U1C7 Outage - 1N11B Feedwater Sensing Line Leak Repair
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TSR 09-0002, U2C15 Outage Torus Snubber Bay 8 &11
TSR 09-0003, Unit 2 Reactor Building 664' elevation, Refuel Floor, R-9 and S-Line South of the SFSP
TSR 09-0004, U2C15 Outage Drywell "Appendix A"
TSR 09-0006, U2C15 Outage In-Vessel Inspection Platform
TSR 09-0007, Unit 2 "B" & "D" RHR HEX Inlet Piping and "B & D" RHR HEX Bottom Head Internal area
TSR 09-0009, Unit 2 SE Quad 519' And 541' Crosstie Piping
TSR 09-0010, U2C15 U-2 RXB O/S Clean Room SE Corner, 565' At Column R-11/R-12 and Between T and U- Lines
TSR 09-0011, Unit 2 RXB 565' R-12 U-Line RHR Vent Line

TSR 09-0012, U2C15 Outage Drywell Activities
TSR 09-0013, U2C15 Unit 2 Drywell, 563' elevation, A,B,C & D Main steam line in support MSIV maintenance
TSR 09-0014, Unit 2 FCV-2-69-2
TSR 09-0022, Unit 0 Refuel Floor Overhead Crane
TSR 09-0024, U2C15 U-2 RXB 565' R-14/R Line (RHR Control Point)
TSR 09-0025, U2C15 "D" MSIV in the U2 Steam Tunnel
TSR 09-0034, U2C15 Junction of 3" and 4" Demin 2B Precoat Inlet Line (Photo attached)
TSR 09-0048, U2C15 FPC Valves 2-DRV-078-559 & 560

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PER 153752 Documenting Source Term Reduction Efforts Being Deferred
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Procedures, Manuals, and Guides
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Procedures, Manuals, and Guides
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0-OI-77B, Floor Drain Collector System Processing, Rev. 57
0-OI-77C, Radwaste Filter and Demineralizer System, Rev. 39
0-OI-77D, Backwash Receivers and Phase Separators System, Rev. 34
0-OI-77E, Solid Radwaste, Rev. 41
0-OI-77F, Thermex Operation, Rev. 20
0-OI-77G, Duratek Procedure FO-OP-32, Set Up and Operating Procedure for the RDS-1000 Unit at TVA Browns Ferry, Rev. 1
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Radioactive Material Shipment Manual (RMSM), Volume I – Information, Rev. 38
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RCI-15.3, Radiological Safety Committee, Rev. 25

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RWTP-100, Radioactive Material/Waste Shipments, Rev. 5
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RWTP-102, Use of Casks, Rev. 1
SPP 3.1, Corrective Action Program, Rev. 15
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Cask Book for Model CNS 8-120B, Rev. 30
Safety Analysis Report 14-195H Shipping Cask, Rev. 2
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Shipment No. 080320, Box Containing GE Equipment, 03/31/08
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Shipment No. 081210, Drum Containing10 CFR 61 Samples, 12/29/08
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Shipment No. 090511, Cask of Dewatered Condensate Resin, 05/13/09
Shipment No. 090519, Sealand of Plant Waste, 05/20/09
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Shipment No. 090521, Sealand of Contaminated Plant Protective Clothing, 05/20/09

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PER 146000, Accident involving radioactive material shipment
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Browns Ferry Units 1/2/3 Mitigating Systems Performance Indicator MS05 through Q1/2009
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LER 05000296/2007-004, Manual Isolation of High Pressure Core Injection due to Steam Leak
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PER 142319, WRM-2 unit LHRA electronic dosimeter retrieval

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SPP-3.1, Corrective Action Program, Rev. 14
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PIDP-12, Integrated Trend Review, Rev. 1
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PER 156229, Inaccurate Unit 2 Risk Condition Communicated for 4KV SD BD C
PER 157170, 1B CRD Pump and 161KV Grid Status Risk Information Inaccurate
PER 159157, POD Risk Status Page Not Updated Accurately
PER 160093, Lack of Protected Equipment Postings for Unit 2 HPCI OOS
PER 160566, Trend PER on Plant Risk
PER 160698, Mode Transition from Mode 4/5 to 3 Without Changing from ORAM to Sentinel
PER 163604, SPP-7.1 Delays in Issuance
PER 167965, Communication of Plant Risk
PER 169790, SPP-7.1 and 7.3 Lack of Change Management
PER 169925, EOOS (CAFTA Model) Coverage Gap
PER 170012, SPP-7.3 Lack of Change Management and Communications
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SPP-7.3, Work Activity Risk Management Process, Rev. 0
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Section 4OA3: Event Follow-up

1-SR-3.3.5.1.6(CS I), Core Spray System Logic Functional Test Loop I, Rev. 4
BFN Unit 1 Technical Specifications Section 3.3.5.1, ECCS Instrumentation
BFN Unit 1 Technical Specifications Section 3.5.1, ECCS - Operating
BFN USFAR Section 6.4.3, Core Spray System
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PER 166487, Unplanned Entry into TS 3.0.3

PER 163815, Unit 1 Generator Trip from Isophase Bus Ground
PER 163815 Root Cause Analysis Report
PER 158940, Unit 1 Bus Duct Cooling Excessive Moisture
PER 163680, Unit 2 Reactor Scram - Stator Coil Temp Hi
PER 163680 Root Cause Analysis Report

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management 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