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

January 30, 2009

Mr. William R. Campbell, Jr.
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2008005, 05000260/2008005 AND 05000296/2008005

Dear Mr. Campbell:

On December 31, 2008, 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 January 9, 2009, with Mr. Rusty West and other members of your staff.

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

Based on the results of this inspection, one NRC-identified finding of very low safety significance (Green) was identified. This finding was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you wish to contest this non-cited violation, 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 20555-001; 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 the Browns Ferry Nuclear Plant.

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/***

Heather J. Gepford, Acting 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 05000259/2008005, 05000260/2008005 and 05000296/2008005
w/Attachment: Supplemental Information

cc w/encl. (See page 3)

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/*

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X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE

ADAMS: X Yes ACCESSION NUMBER:________________________

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| OFFICE | RII:DRP | RII:DRP | RII:DRP | RII:DRP | RII:DRP | RII:DRS |
| SIGNATURE | /RA/ | /RA By E-mail/ | /RA By E-mail/ | /RA By E-mail/ | /RA By E-mail/ | /RA By E-mail/ |
| NAME | JBaptist | CPeabody | LSuggs | CStancil | TRoss | RCarrion |
| DATE | 1/29/09 | 1/29/09 | 1/29/09 | 1/27/09 | 1/29/09 | 1/27/09 |
| E-MAIL COPY? | YES NO | YES NO | YES NO | YES NO | YES NO | YES NO |
| OFFICE | RII:DRS | RII:DRS | RII:DRP | RII:DRP | RII:DRP | RII:DRP |
| SIGNATURE | /RA By E-mail/ | /Via Telecom/ | /RA/ | | | |
| NAME | MBates | SRose | HGepford | | | |
| DATE | 1/27/09 | 1/29/09 | 1/30/09 | | | |
| E-MAIL COPY? | YES NO | YES NO | YES NO | YES NO | YES NO | YES NO |

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Letter to William R. Campbell, Jr. from Heather J. Gepford dated January 30, 2009.

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2008005, 05000260/2008005 AND 05000296/2008005

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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.: | 05000259/2008005, 05000260/2008005 and 05000296/2008005 |
| Licensee: | Tennessee Valley Authority (TVA) |
| Facility: | Browns Ferry Nuclear Plant, Units 1, 2, and 3 |
| Location: | Corner of Shaw and Nuclear Plant Roads<br>Athens, AL 35611 |
| Dates: | October 1, 2008 through December 31, 2008 |
| Inspectors: | T. Ross, Senior Resident Inspector<br>C. Stancil, Resident Inspector<br>L. Suggs, Reactor Inspector<br>C. Peabody, Reactor inspector<br>J. Baptist, Senior Project Engineer<br>R. Carrion, Senior Reactor Inspector (Section 1R08)<br>M. Bates, Senior Operations Engineer (Section 1R11)<br>S. Rose, Senior Reactor Inspector (Section 1R11) |
| Approved by: | Heather J. Gepford, Acting Chief<br>Reactor Projects Branch 6<br>Division of Reactor Projects |

Enclosure

## SUMMARY OF FINDINGS

IR 05000259/2008005, 05000260/2008005 and 05000296/2008005; 10/01/2008 – 12/31/2008; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Licensed Operator Requalification Program.

The report covered a three month period of inspection by resident inspectors and reactor inspectors from the region. One non-cited violation (NCV) was 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: Mitigating Systems

Green. The inspectors identified a non-cited violation of 10 CFR 55.49 for engaging in an activity that compromised, or would have compromised but for detection by the inspectors, the integrity of examinations required by 10 CFR 55.59 that were administered in 2007 and that were planned to be administered in 2008. The examination compromise would have affected the equitable and consistent administration of the operational portion of the requalification annual examination. The inspectors identified that three job performance measures (JPM) sets administered in 2007 contained an unacceptable number of JPMs that had previously been administered during that same examination cycle. The inspectors also identified that the JPMs scheduled to be performed in the last three weeks of the 2008 requalification examination had all been previously administered in the first three weeks of the 2008 requalification examination. When notified of the examination schedule overlap issue, the licensee changed the examination schedule to prevent the overlap issue in 2008 and entered the problem into their corrective action program as problem evaluation report 158635.

This finding is more than minor because if left uncorrected, it could become a more significant safety concern, in that, licensed operators would not be adequately tested to ensure an acceptable knowledge level for performing licensed duties. Using the Licensed Operator Requalification Significance Determination Process, this finding was determined to be of very low safety significance (Green) because the performance deficiency was immediately corrected upon discovery. The cause of the finding was that the licensee did not comply with requirements of TRN-11.10, Annual Requalification Examination Development and Implementation. The finding was related to the cross-cutting aspect of procedural compliance of the work control component of the cross-cutting area of Human Performance (H.4(b)). (Section 1R11.2)

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 began this report period at full Rated Thermal Power (RTP). On October 25, 2008, the unit was shutdown to conduct the Unit 1 Cycle 7 (U1C7) refueling outage. The unit was restarted November 30, 2008, and the main turbine generator was placed on the grid two days later. On December 3, 2008, an unplanned downpower was conducted from approximately 38 percent RTP to 22 percent RTP in order to take the main turbine generator offline and repair an unisolable steam leak downstream of Control Valve #1. Following the repairs, the unit was returned to the grid on December 4, 2008, and achieved full power on December 7, 2008. On December 11, 2008, an unplanned downpower was conducted on Unit 1 to 50 percent RTP to repair an electrical ground on the actuator of the Reactor Core Isolation Cooling (RCIC) System outboard steam admission valve (FCV-71-03) located in the Main Steam Tunnel. The unit was returned to full RTP on December 12, 2008 and remained at full RTP for the remainder of the report period.

Unit 2 operated at essentially full RTP the entire report period except for an automatic scram and a planned downpower. On October 4, 2008, an automatic reactor scram from 100 percent RTP occurred on Unit 2 due to a full load rejection when the main turbine generator tripped. The generator tripped as a result of the loss of main turbine generator excitation due to a loose wire in the automatic voltage regulator circuit and oxidized contacts on a relay in the manual voltage regulator circuit. The unit was restarted on October 7, 2008, and returned to full RTP on October 10, 2008. On December 13, 2008, a planned downpower to approximately 70 percent RTP was conducted to perform a control rod sequence exchange, scram time testing, turbine valve testing, and main condenser waterbox cleaning. The unit was returned to full RTP on December 12, 2008.

Unit 3 operated at essentially full RTP the entire report period except for one planned downpower. On November 15, 2008, a planned downpower to approximately 70 percent RTP was conducted to perform a control rod sequence exchange, scram time testing and main condenser waterbox cleaning. The unit was returned to full RTP on November 16, 2008.

## 1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

### 1R01 Adverse Weather Protection

#### .1 Readiness for Seasonal Extreme Weather Conditions (Cold Weather Preparation)

a. Inspection Scope

Prior to and during the onset of cold weather conditions, the inspectors performed a review of the licensee's implementation of licensee procedure 0-GOI-200-1, Freeze Protection Inspection, including applicable checklists: Attachment #1, Freeze Protection Annual Checklist; Attachment #2, Freeze Protection Operational Checklist; and as applicable, Attachments #3 through #12, Freeze Protection Daily Log Sheets for individual watch stations. The inspectors also reviewed the Freeze Protection

Printout (PA-304) and discussed implementation of 0-GOI-200-1 with responsible Operations personnel and management. Furthermore, to verify that affected systems and components were properly configured and protected, the inspectors conducted walkdowns of potentially affected risk significant equipment located in the residual heat removal service water (RHRSW) and emergency equipment cooling water (EECW) pump rooms, both emergency diesel generator (EDG) buildings, intake structure, all reactor building ventilation intake plenums, auxiliary decay heat removal (ADHR) system, standby gas treatment (SBGT) building, and the outside tank area.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

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

- Unit 1 RCIC System
- Unit 2 High Pressure Coolant Injection (HPCI) System
- Unit 1 and 2 EDGs A and B

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 four fire zones (FZ) listed below. Selected FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure SPP-10.9. Furthermore, the inspectors reviewed applicable portions of the Site Fire Hazards Analysis Volumes 1 and 2 and Pre-Fire Plan drawings to verify that the necessary fire fighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, were in place.

- Unit 1 Reactor Building EL 519’ to 565’, West (FZ 1-1)
- Unit 1 Reactor Building EL 519’ to 565’, East (FZ 1-2)
- Unit 1 Reactor Building EL 593’, North of Column Line R (FZ 1-3)
- Unit 1 Reactor Building EL 593’, South of Column Line Q (FZ 1-4)

b. Findings

No findings of significance were identified.

.2 Annual Drill Observation

a. Inspection Scope

On October 22, 2008, the inspectors witnessed an unannounced fire drill in the Unit 3 Turbine Building at the Main Turbine Lube Oil Tank. The inspectors assessed fire alarm effectiveness; response time for notifying and assembling the fire brigade; the selection, placement, and use of fire fighting equipment; use of personnel fire protective clothing and equipment (e.g., turnout gear, self-contained breathing apparatus); communications between incident commander and control room; teamwork; and fire fighting strategies. The inspectors also attended the post-drill critique to assess the licensee’s ability to review fire brigade performance and identify areas for improvement. Following the critique, the inspectors compared their findings with the licensee’s observations.

b. Findings

No findings of significance were identified.

1R06 Internal Flood Protection Measures

a. Inspection Scope

The inspectors performed a review of the flood protection measures for the RHRSW pump rooms and cable tunnel. The inspectors reviewed plant design features and measures intended to protect the plant and its safety-related equipment from internal flooding events, as described in the following documents: UFSAR, General Design Criteria, and the Probabilistic Safety Assessment Internal Flooding Notebook. The inspectors performed walkdowns of risk-significant areas, susceptible systems and equipment that included the A, B, C, and D RHRSW Rooms, and associated power and instrument cabling running between the Power Block and the Intake Structure, to verify the condition of flood-mitigation features such as area level switches, room sumps and sump pumps, conduit seals, and instrument racks that might be subjected to flood conditions. The inspectors also reviewed a sampling of the licensee’s corrective action documents with respect to flood-related items to verify that problems were being identified and corrected. Furthermore, the inspectors reviewed selected completed preventive maintenance procedures, work orders (WO), and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

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

a. Inspection Scope

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) pressure boundary and risk-significant piping boundaries during the U1C7 refueling outage. The inspectors' activities consisted of an on-site review of nondestructive examination (NDE) and welding activities to evaluate compliance with TS and the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections XI and V (Code of record: 2001 Edition through the 2003 Addenda), for Class 1, 2, and 3 systems; 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 Unit 1, this was the first outage of the first period of the second interval. The inspectors also reviewed a sample of inspection activities associated with components that were outside the scope of ASME Section XI requirements which were performed in accordance with commitments to follow industry guidance documents, such as the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

The inspectors reviewed NDE activities, specifically including examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, and calibration reports for compliance with requirements of ASME Section V, ASME Section XI, BWRVIP documents, and other industry standards for the following examinations:

- Phased Array UT examination of weld RHR-1-014-001, ASME Class 1, Residual Heat Removal (RHR) System, 24-inch diameter pipe-to-elbow (Section XI ISI) - Direct Observation (Report # R-101)

- PT examination of weld DRHR-1-11, ASME Class 2, RHR System, 24-inch diameter pipe-to-valve (Section XI ISI) - Direct Observation

The inspectors also reviewed NDE reports of visual (VT) examinations, ultrasonic (UT) examinations, and magnetic particle (MT) examinations as described below:

| VT Report | Component |
|---|---|
| R-001 (VT-3) | 1-47B465-497, Snubber in the Reactor Water Recirculation (RWR) System |

| | |
|---|---|
| R-002 (VT-2) | RHRSW System Leak Test (on "B" Header and "D" Header) |
| R-024 (VT-3) | 1-47B450-4, Rigid Strut in the RHRSW System |
| R-053 (VT-3) | 1-47B400-104, Hydraulic Snubber in the Main Steam System |

| UT Report | Component |
|---|---|
| R-027 | RHRG-1-07-C, RHR heat exchanger "C" Shell to Flange Weld |
| R-067 | Mixing Tee in the Control Rod Drive System |
| R-081 | Reactor Water Cleanup 1-005-005, Valve to Elbow Weld |

| MT Report | Component |
|---|---|
| R-075 | 1-47B400-94-IA, Main Steam Anchor |

The inspectors conducted a Unit 1 walk-down of multiple drywell elevations to assess, in general, the material condition of structures, systems, and components (SSCs) including leaks from bolted connections, coating integrity, cleanliness, hangers and supports, etc. In addition, the inspectors walked down the exterior of the torus and reviewed the licensee's operating experience assessment for issues associated with NRC Information Notice 2006-01, "Torus Cracking in a BWR Mark I Containment."

The inspectors completed 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 problems and had implemented appropriate corrective actions. The inspectors' review included confirmation that the licensee had an adequate threshold for identifying issues. Through interviews with licensee staff and review of corrective action documents, the inspectors evaluated the licensee's threshold for identifying lessons learned from industry issues related to ASME Section XI. The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

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 (BWRVIP):

- VT-3 of Jet Pump sensing line clamps
- EVT-1 of circumferential crack indication in the backing ring for the DF-3 weld (of Jet Pump 19) to determine if indication had increased in length
- VT-1 of Core Spray T-box
- VT-3 of Core Plate hold-down bolts
- VT-1 of Steam Dryer drain channel #2 horizontal weld (DC-2-H2)
- VT-3 of Core Shroud replacement access hole covers

The inspectors also observed activities related to the modifications of the Unit 1 steam dryer. The original tie bars and gusset plates had been re-designed and were being replaced due to operational experience at other plants which found the original design to be susceptible to structural cracking. The replacement activities included underwater welding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On October 20, 2008, the inspectors witnessed three Licensed Operator Requalification (LOR) Program annual simulator examinations of one crew. During each exam, the senior reactor operator and reactor operator positions were rotated, except for the shift manager. The exams witnessed by the inspectors were the 2008 LOR-Exam-23, 29, and 42.

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

b. Findings

No findings of significance were identified.

.2 Biennial Review

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of December 1 - 5, 2008, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR Part 55, Operators' Licenses. The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." The inspectors also evaluated the licensee's simulator facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1985, American National Standard for Nuclear Power Plant Simulators for use in Operator Training and Examination. The inspectors observed two crews during the performance of the operating tests. Documentation reviewed included written examinations, job performance measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records and performance test records, operator feedback records, licensed operator qualification records, remediation plans, watchstanding records, and medical records. These records were inspected using the criteria listed in Inspection Procedure 71111.11, Licensed Operator Requalification Program. Documents reviewed are listed in the Attachment to this report.

b. Findings

Introduction: The inspectors identified a non-cited violation of 10 CFR 55.49 for engaging in an activity that would have affected the equitable and consistent administration of the operational portion of the requalification annual examination.

Description: The licensee is required to administer one set of five JPMs to each licensed operator as part of each licensed operator's annual operating examination. On December 3, 2008, while reviewing the licensed operator requalification JPM schedule, the inspectors identified that the licensee was planning on repeating JPMs that were administered earlier in the 2008 examination cycle. The inspectors identified that the JPMs scheduled to be administered the last three weeks of the 2008 requalification schedule had all been previously administered in the first three weeks of the 2008 requalification schedule. The inspectors identified that the licensee failed to follow their procedural requirements to ensure proper selection of JPMs for examination administration. The licensee's training procedure, TRN-11.10, Annual Requalification Examination Development and Implementation, Revision 12, states in Section 3.16, Selecting Job Performance Measures, Step A.5, "At least 50 percent of the JPMs used in any one set of 5 SHALL be different from JPMs used in any other week of the exam cycle."

Contrary to the above, the licensee's 2008 examination schedule reflected that eight JPM sets were scheduled to repeat four of the five JPMs from a previous week in the cycle and would have resulted in only 20 percent of the JPMs being different from JPMs used in any one previous week of the examination cycle. The inspectors notified the licensee of the examination schedule overlap issue and the licensee subsequently changed the schedule to prevent the overlap issue and entered the problem into their corrective action program as Problem Evaluation Report (PER) 158635. Additionally, the inspectors reviewed the 2007 operating examination administration and identified that three JPM sets had violated the above requirement specified in Step A.5 of Section 3.16 of TRN-11.10. The inspectors compared the pass/fail results for the JPMs initially administered in those three sets in the 2007 examination cycle, prior to the overlap issue being present, to the pass/fail rates for the same JPMs when they were administered again in that same 2007 examination cycle. However, due to the high pass rate on the subject JPMs when they were initially administered, no conclusions could be made regarding an increasing pass rate.

Analysis: The failure to follow procedure Step A.5 of Section 3.16 of TRN-11.10 led to the development of multiple JPM sets that would have compromised the integrity of the individual operating examination for 2008 and could have compromised the 2007 operating examination, which was a performance deficiency. This finding is more than minor because, if left uncorrected, it would become a more significant safety concern in that licensed operators would not be adequately tested to ensure an acceptable knowledge level for performing licensed duties. It affected the human performance attribute of the Mitigating Systems cornerstone because licensed operator response to initiating events mitigates undesirable consequences. The significance determination was performed in accordance with Manual Chapter 0609, Significance Determination Process, Appendix I, Licensed Operator Requalification Significance Determination Process (SDP). Question 5 in Appendix I, asks if the finding was related to the individual operating test quality, security or operator performance in the walkthrough (generally, job performance measures). The answer to this question was "YES" because the finding was related to the licensee's program for maintaining the integrity of the tests. Question 8 asks if the integrity of the individual operating test had been compromised. The answer to this question was "YES" because the inspectors identified that three of the JPM sets administered in 2007 where not administered in accordance with the licensee's procedures to ensure examination integrity. Additionally, if the licensee had implemented the 2008 examination schedule as originally planned, the examination integrity would have been compromised and therefore, would have affected the equitable and consistent administration of the examination. 10 CFR 55.49 states, in part that the integrity of a test or examination is considered compromised if any activity, regardless of intent, affected, or, but for detection, would have affected the equitable and consistent administration of the test or examination. Question 11 asks if when the compromise was discovered (or should have been discovered), did the licensee take compensatory measures immediately. The answer to this question was "YES" because the licensee changed the schedule to ensure at least 50 percent of the JPMs used in any set of 5 would be different from JPMs used in any one previous week of the examination cycle. Consequently, this finding is characterized as (Green), having very low safety significance.

The finding was directly related to the cross-cutting aspect of procedural compliance of the work practices component of the cross-cutting area of Human Performance. The examination developers did not comply with training procedure requirements to ensure examination integrity was maintained (H.4(b)).

Enforcement: 10 CFR 55.49 states that applicants, licensees, and facility licensees shall not engage in any activity that compromises the integrity of any application, test, or examination required by this part. The integrity of a test or examination is considered compromised if any activity, regardless of intent, affected, or, but for detection, would have affected the equitable and consistent administration of the test or examination. This includes activities related to the preparation and certification of license applications and all activities related to the preparation, administration, and grading of the tests and examinations required by this part. Activities covered by this part include the requirements stated in 10 CFR 55.59, Requalification.

Contrary to the above, the inspectors identified three JPM sets that were administered during the 2007 operating examination cycle and eight JPM sets scheduled to be administered during the 2008 operating examination cycle that violated 10 CFR 55.49. The licensee procedure, TRN-11.10, Step A.5 of Section 3.16, was not implemented during the selection and scheduling of JPM sets to be administered during the 2007 and 2008 requalification examination cycles because more than 50% of the JPMs used in any one set were not different from JPMs used in any other week of the examination cycle. Failure to implement this procedure step compromised the integrity of the examinations in that, if left uncorrected, it would have affected the equitable and consistent administration of the test or examinations. Because this issue is of very low safety significance and has been entered into the licensee's corrective action program as PER 158635, the violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 50-259, 260, 296/2008005-01, Failure to Maintain Requalification Examination Integrity.

## 1R12 Maintenance Effectiveness

### .1 Routine

#### a. Inspection Scope

The inspectors reviewed three specific equipment issues listed below for SSCs within the scope of the Maintenance Rule (MR) (10 CFR 50.65) regarding 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) of the MR; (4) Characterizing reliability issues for performance; (5) Trending key parameters for condition monitoring; (6) Tracking unavailability and function failures; (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. Furthermore, the inspectors reviewed applicable work orders, surveillance records, 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 3 Drywell Equipment Hatch 3-X-1A Leakrate Failure
- RHRSW System Piping Leaks
- Unit 2 RHR Drywell Spray Inboard Isolation Valve (FCV-74-75) Repeat Breaker 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 four 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.

- A EDG, 2A Control Rod Drive (CRD) Pump and C3 EECW Pump out of service (OOS)
- C 4KV Shutdown Board, C EDG, 2A CRD Pump OOS, and 1B 480V Shutdown Board on alternate power supply
- 3B CRD Pump, 3D RHR Pump, C3 EECW Pump and 3B Reactor Protection System (RPS) Motor Generator set OOS
- Loss of the 1B Spent Fuel Pool Cooling (SFPC) Pump with the 1A SFPC Pump and the ADHR System OOS

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the six operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately evaluated TS operability. The inspectors also reviewed applicable sections of the TS and UFSAR to verify that the system or component remained available to perform its intended function.

In addition, where appropriate, the inspectors reviewed licensee procedure NEDP-22, Functional Evaluations, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors examined the implementation of compensatory measures to verify that they achieved the intended purpose and that the measures were adequately controlled. The inspectors also reviewed PERs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- RHRSW Degraded Flow Rate Capabilities (PER 134465)
- Equipment Access Lock Flood Gate Seals Not Installed Nor Staged (PER 154175)
- Units 1, 2, and 3 Standby Diesel Generators (8) Unit Priority Trip Relay Common Cause Failures (PER 156416)
- Unit 1 Lost Welding Rod Evaluation (PER 157270)
- Unit 1 Reactor Vessel Water Level Erratic Indication During Cooldown (PER 155534)
- U1 Jet Pump Set Screw Cracks (PER 156982)

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the three 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, the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and 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, Pre-/Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors reviewed problems associated with PMTs that were identified and entered into the licensee's corrective action program (CAP).

- RHR pumps 1B and 1D per 1-SR-3.5.1.6 (RHR II-COMP) RHR Loop II Comprehensive Pump Test
- Units 1 and 2 C Standby Diesel Generator Unit Priority Retrip Relays 1- and 2-RLY-74-10AK132B per WOs 08-723558-001 and -002, and 1/2-SR-3.3.5.1.6 (A II), Functional Testing of RHR Loop II Automatic Initiation Logic and Injection Valve Opening Pressure Permissive Logic
- Unit 1 RCIC Electronic Governor Replacement per WOs 08-725178-000, 07-762220-000 and 08-714794-000 as well as 1-SI-3.5.3.3(COMP) RCIC Comprehensive Flow Test

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 1 Scheduled Refueling Outage

a. Inspection Scope

From October 25 to December 7, 2008, the inspectors examined critical outage activities of the U1C7 refueling outage to verify that they were conducted in accordance with TS, applicable procedures, and the licensee's outage risk assessment and management plans. Some of the more significant inspection activities conducted by the inspectors are described below.

Outage Risk Assessment

Prior to the scheduled 25-day U1C7 refueling outage, that began on October 25, the inspectors 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. During the outage, the inspectors also reviewed the daily U1C7 Refueling Outage Reports, including the Outage Risk Assessment Management (ORAM) results and Safety Function Status, and regularly attended the twice-a-day outage status meetings (i.e., STORM meetings). These reviews were compared to the requirements in licensee procedure SPP-7.2, Outage Management, and TS. 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 1 in accordance with licensee procedures 1-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; and 1-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 1-OI-74, Residual Heat Removal System; 1-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Alternate Decay Heat Removal System Failures; and conducted main control room panel and in-plant walkdowns of the decay heat removal 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 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 inspections of outage activities completed by the inspectors were as follows:

- Walked down selected safety-related equipment clearance orders (i.e., tag order 1-TO-2008-003, section 1-074-0004A for 1-CKV-074-0559A, RHR Pump 1A Discharge Check Valve; and, 1-TO-2008-003, section 1-023-002 for RHR HX 1D (RHRSW Side)
- Verified Reactor Coolant System (RCS) inventory controls
- Verified electrical systems availability and alignment
- Monitored critical plant parameters (e.g., RCS pressure, level, flow, and temperature), and verified 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 routine tours of the control room, 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 1-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, the inspectors witnessed fuel handling operations during the Unit 1 reactor core fuel shuffle 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.

Drywell and Torus Closeout

On November 10, the inspectors reviewed the licensee's conduct of the applicable portions of procedure 1-GOI-200-2, Primary Containment Initial Entry and Closeout, for torus closeout (Checklist 9) and performed an independent detailed closeout inspection of the Unit 1 torus.

On November 30, the inspectors reviewed the licensee's conduct of procedure 1-GOI-200-2 for drywell closeout and performed an independent detailed closeout inspection of the Unit 1 drywell.

Restart Activities

The inspectors specifically conducted the following:

- Witnessed Unit 1 approach to criticality in accordance with procedure 1-GOI-100-1A, Unit Startup

- Observed portions of reactor coolant heatup/pressurization to rated temperature and pressure per 1-SR-3.4.9.1, Reactor Heatup and Cooldown Rate Monitoring
- Observed power ascension of Unit 1 in accordance with procedures 1-GOI-100-1A; 1-GOI-100-12, Power Maneuvering; and the applicable Reactivity Control Plan
- Witnessed portions of the control rod scram time testing at power per procedure 1-SR-3.1.4.1, Scram Insertion Times

Corrective Action Program

The inspectors reviewed PERs generated during U1C7 and attended corrective action review board (CARB) 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 and/or verified for adequacy and 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 six 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:

- 3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure

Routine Surveillance Tests:

- 1-SR-3.3.3.2.1(75 II), Backup Control Panel Testing
- 1-SR-3.7.5.1, Turbine Bypass Valve Cycling
- 1-SR-3.8.1.9(A), Diesel Generator A Emergency Unit 1 Load Acceptance Test
- 1-SR-3.4.3.2, Main Steam Relief Valve Manual Cycle Test

Containment Isolation Valve Tests:

- 2-SR-3.6.1.3.5(76 II), $H_2O_2$ System Isolation Valve Operability Test (Division II)

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On October 1, 2008, 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 to identify any weaknesses and deficiencies in classification and notification activities. The inspectors observed emergency response operations in the Unit 2 simulated control room and the Technical Support Center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure and other applicable Emergency Plan Implementing Procedures. 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 and entering them into the CAP as appropriate.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Initiating Events

Unplanned Scrams, Unplanned Scrams with Complications, and Unplanned Power Changes

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following PIs, including procedure SPP-3.4, Performance Indicator for NRC Reactor Oversight Process for Compiling and Reporting PIs to the NRC. The inspectors examined the licensee's PI data for the specific PIs listed below for the fourth quarter of 2007 through the third quarter of 2008. The inspectors reviewed the licensee's data and graphical representations as reported to the NRC to verify that the data was correctly reported. The inspectors also validated this data against relevant licensee records (e.g., PERs, Daily Operator Logs, Plan of the Day, Licensee Event Reports, etc.), and assessed any reported problems regarding implementation of the PI program.

Furthermore, the inspectors met with responsible plant personnel to discuss and go over licensee records to verify that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors also used the Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to ensure that industry reporting guidelines were appropriately applied.

- Unit 1 Unplanned Scrams
- Unit 2 Unplanned Scrams
- Unit 3 Unplanned Scrams
- Unit 1 Unplanned Scrams with Complications
- Unit 2 Unplanned Scrams with Complications
- Unit 3 Unplanned Scrams with Complications
- Unit 1 Unplanned Power Changes
- Unit 2 Unplanned Power Changes
- Unit 3 Unplanned Power Changes

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 and periodically attending daily PER screening and review team meetings.

.2 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review 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. This review also included issues documented outside the normal CAP, such as in system health reports, corrective maintenance WOs, component status reports, site excellence meeting reports, and MR assessments. The inspectors' review nominally considered the six-month period of July 2008 through December 2008, 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. In particular, the inspectors reviewed the licensee's Integrated Trend Review (ITR) program and the implementation of the program. Inspectors also interviewed responsible licensee management.

As mentioned in the Annual Assessment Letter - Browns Ferry Nuclear Plant (NRC Inspection Report 05000259/2008001, 05000260/2008001, and 05000296/2008001), the inspectors continued to monitor licensee efforts to ensure effective and timely corrective action with respect to the cross-cutting aspect (P.1.d), appropriate and timely corrective action.

b. Findings and Observations

No findings of significance were identified. In general, the licensee has identified trends and has appropriately addressed the trends in their CAP. However, the inspectors identified three adverse or potentially adverse trends in the areas of procedure use and adherence, plant risk assessment, and maintenance rule (10 CFR 50.65) program implementation. These adverse or potentially adverse trends were associated with violations of minor significance that are not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

- Eight examples of procedure use and adherence issues were identified, involving individuals not performing an adequate verification of certain procedure steps. This resulted in the specified procedure steps not being performed as required. These examples were discussed with the licensee, and the inspectors evaluated the consequences of each issue. For each of the individual issues, the licensee generated PERs. In addition, the licensee generated PER 159472 to document an adverse trend in the area of procedural verification of work activities.

- The inspectors discussed several instances in which risk assessments from ORAM, the shutdown plant operational risk assessment model, and SENTINEL, the operating plant computer based risk management model, did not accurately reflect the actual plant equipment configuration. The licensee's preliminary review of the issues indicated that lack of ownership within shift Operations and communications errors between station departments were key contributors to these issues. In each instance, the licensee generated PERs and re-evaluated the plant risk, based on actual plant equipment configuration, and found no significant change to actual plant risk. The inspector's observations were discussed with the licensee and the inspectors reviewed the licensee's re-evaluated risk results. Additionally, the licensee generated PER 160566 to evaluate the existence of a potential adverse trend in the area of plant risk management and to develop any necessary corrective actions.

- Licensee procedures establish guidelines for the review and evaluation of the reliability and availability of SSCs that are scoped in the MR program. The licensee's guidelines state that issues affecting SSCs in the maintenance rule program shall be evaluated within a specific time frame (i.e., 60 days) to ensure issues are addressed in a timely manner commensurate with their safety significance. This evaluation consists of a cause determination evaluation (CDE) developed by the system engineer to initiate the MR process. However, during inspector reviews of various reliability and unavailability issues related to SSCs scoped in the MR program, the inspectors observed that the licensee was not meeting their own procedural guidelines with respect to the established time frame for developing an CDE. In response to the inspectors' observations, the licensee conducted a review of their implementation of the program guidelines and found that 17 CDE's, since September 2007, had exceeded their procedural completion guidelines. Of these 17 CDE's, seven had not been initiated. A similar issue regarding six overdue CDE's was previously identified by the licensee in June 2007 as documented by PER 126875. The licensee acknowledged that problems continued to exist regarding timeliness of initiating and completing CDE's. In addition to CDE's not meeting licensee timeliness expectations, the inspectors identified that several 10 Point Plans for establishing MR goals and monitoring of SSCs (e.g., Unit 3 Equipment Hatch and Unit 2 FCV 2-74-75) that had already been classified as 10 CFR 50.65(a)(1), had been delayed for an extended period of time (i.e., many months). The licensee also acknowledged these 10 Point Plans did not meet their expectations for timeliness. The inspectors evaluated the delayed actions and issues associated with the aforementioned CDE's and 10 Point Plans and determined that there were no regulatory findings of significance. These issues were discussed with the licensee regarding an adverse trend and the licensee initiated PER 152007 to resolve these identified concerns.

While the licensee initiated corrective actions to verify that each individual issue was addressed, the licensee did not identify that the above issues constituted apparent trends, thus necessitating a trend PER in order to capture all appropriate corrective actions.

The inspectors also determined that the licensee, in general, had not fully implemented their ITR program. This program established guidance for conducting departmental and site trending at least once every semi-annual period. However, inspectors noted that site ITR summaries were not issued for the first or second semi-annual periods for 2008, and departmental reports were not prepared for the second semi-annual period (July to December 2008). The licensee self-initiated PER 150254 for the failure to complete the first period ITR and initiated PER 151684 to document corrective actions for failure to meet procedure requirements and quality/content of trend reviews. The licensee further stated that existing performance improvement department procedures were being revised and new procedures developed to provide additional guidance on conducting departmental and site trending.

.3 Focused Annual Sample Review – Preventive Maintenance (PM) Program

a. Inspection Scope

In February 2008, the licensee instituted the Turnaround Plan to Excellence for the Browns Ferry Nuclear Plant. One of the major focus areas of this Plan was "Equipment Reliability." In the area of Equipment Reliability, the inspectors reviewed the licensee's corrective action plans to address "Preventive Maintenance Program Issues". The inspectors specifically reviewed the licensee's root cause analysis (RCA) and corrective actions associated with PER 153450, Preventive Maintenance Program, including licensee compliance with their RCA program guidance. The RCA evaluated common cause critical (important to plant safety) component failures and weaknesses in the current PM program, including PM deferral justifications. Additionally, the inspectors reviewed other related PERs and corrective actions resulting from longstanding PM program issues. Furthermore, the inspectors interviewed responsible engineers, maintenance and scheduling management, scheduling and planning personnel, and Turnaround Plan management.

b. Findings and Observations

No findings of significance were identified. However, the inspectors documented the following observations:

Licensee PERs were initiated, starting in 2006, concerning PM deferral issues, including trending, technical justifications, and escalation criteria. These 2006 corrective actions resulted in minor administrative changes; however, critical component failures attributed to the ineffective implementation of the PM program continued to occur. Near the end of 2006, the licensee identified equipment reliability issues as a result of PM activities not being identified, planned, or implemented when scheduled and subsequent deferrals not including adequate bases and historical evaluations. To address these issues, PER 116524 was initiated. A similar PM program review by the licensee in 2008 resulted in identification of similar issues, which were addressed by the RCA of PER 153450. Coincident with the 2008 PM program review was an audit by the site's nuclear assurance group which also resulted in two PERs, 152795 and 152846, on the same PM issues.

The licensee identified two root causes from the PER 153450 RCA: 1) effective equipment reliability strategies had not been defined and implemented for critical components; and 2) the licensee's PM procedural guidance did not have sufficient guidance, details and examples to develop adequate technical justifications for PM deferrals.

The inspectors reviewed the RCA of PER 153450 and determined that the conclusions appeared to be reasonable and appropriate according to the licensee's RCA guidance. However, the inspectors observed that the licensee's RCA for PER 153450 was delayed and that PM program corrective actions to date have not been effective in reducing the number of critical component failures. These PM issues have been captured in PER 153435, Work Schedule Adherence, which will require a RCA.

.4 Focused Annual Sample Review – Corrective Actions for CAP Trend

a. Inspection Scope

The inspectors reviewed the specific corrective actions and common cause analysis for PER 151140, Potential Negative Trend in the Cross Cutting Program Corrective Action Program. This PER was initiated to evaluate and address the identified causes of a potential negative trend in the cross-cutting component of Corrective Action Program in the aspects of thorough evaluation of identified problems (P.1.c) and corrective action program (P.1.a). The common causes were determined to be:

- The corrective action program procedures lack sufficient detail and/or guidance for proper implementation of the program in the areas of Root Cause Evaluations, Apparent Cause Evaluations, and Extent of Condition/Extent of Cause Evaluations.
- Management oversight and reinforcement of standards with respect to the Corrective Action Program was less than adequate.
- The current corrective action program training materials are deficient in the areas of corrective action program key concepts, program requirements and program tools.

b. Findings and Observations

No findings of significance were identified. The inspectors reviewed and verified the licensee's corrective action plan, common cause analysis, extent of condition, and the corrective actions instituted to address the identified causes. Corrective actions were implemented and/or scheduled to address all the identified causes. Generally, those actions included more stringent requirements for fundamental program aspects, more detailed procedural guidance, revisions to the current CAP processes, and heightened accountability and responsibility for management oversight and reinforcement, as well as development of additional improvements to existing training materials for those involved in CAP screening and evaluation.

## 4OA3 Event Follow-up

### .1 Unit 2 Automatic Reactor Scram

#### a. Inspection Scope

On October 4, 2008, the Unit 2 reactor automatically scrammed from 100 percent power due to a main turbine generator load reject signal. Prior to the Unit 2 scram, control room operators had observed voltage fluctuations on the main turbine generator, to which they responded by placing the main turbine generator voltage regulator in a manual mode of operation. However, immediately after the main turbine generator voltage regulator was placed in manual, the main turbine generator experienced an unexpected loss of excitation resulting in a trip of the main turbine generator. The resident inspectors responded to the control room and verified that the unit was in a stable Mode 3 (Hot Shutdown) condition. The inspectors also confirmed that all safety-related mitigating systems and automatic functions operated properly. Furthermore, the inspectors evaluated safety equipment and operator performance before and after the event by examining existing plant parameters, strip charts, plant computer historical data displays, operator logs, and the critical parameter trend charts in the post-trip report. The inspectors also interviewed responsible on-shift Operations personnel, examined the implementation of applicable alarm response procedures (ARPs), AOIs, and EOIs, particularly 1-AOI-100-1, Reactor Scram. Furthermore, the inspectors reviewed and verified that the NRC required notifications were made in accordance with 10 CFR 50.72.

#### b. Findings

No significant findings were identified.

### .2 (Closed) LER 05000259/2008001-00, Loss of Safety Function Reactor Zone Exhaust Dampers Failed to Close

#### a. Inspection Scope

The inspectors reviewed the subject LER dated November 3, 2008, and the applicable PER 151814, including the associated apparent cause determination and corrective action plans. The inspectors also reviewed related PERs 45849 and 152333.

On September 3, during routine surveillance testing of the 1A1 and 1A2 RPS Circuit Protectors, the Unit 1 reactor operator identified that the inboard and outboard Unit 1 Reactor Zone Exhaust secondary containment isolation dampers (1-DMP-64-42 and 43) had failed to close. These dampers were expected to close automatically as a consequence of the Group 6 Primary Containment Isolation Signal (PCIS) generated by the surveillance test. Both dampers were subsequently declared inoperable and determined by the licensee to constitute a safety system functional failure (SSFF). The dampers were immediately closed and deactivated to comply with TS 3.6.4.2. The cause of the stuck open dampers was determined to be use of a silicone based lubricant by the manufacturer during assembly of the specific ASCO model solenoid valves used to control 1-DMP-64-42 and 43. Since these solenoids were normally energized, the heat caused the lubricant to bake and become sticky. This problem was previously identified by the licensee in March 2002 (PER 45849) when Unit 2 Reactor Zone Supply secondary containment isolation damper 2-DMP-64-14 failed for the same cause. The root cause determination and extent of condition at that time specifically identified that other Unit 1, 2, and 3 damper solenoid valves, including 1-DMP-64-42 and 43, were susceptible to the same failure mechanism. However, the corrective actions of PER 45849 only required changing out the solenoid valves for Units 2 and 3. This PER inadvertently omitted changing out the Unit 1 solenoid valves. To address the inadequate corrective action plan that was originally developed for the extent of conditions identified by PER 45849, the licensee initiated PER 152333.

b. Findings

Since this SSFF of secondary containment isolation was discovered during surveillance testing this LER was dispositioned as an NCV in Section 4OA7 of this report. This LER is considered closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No significant findings were identified.

.2 Independent Spent Fuel Storage Installation Operations Inspection and 10 CFR 72.48 Evaluations Review

a. Inspection Scope

The inspectors examined routine performance of normal independent spent fuel storage installation (ISFSI) operations activities. [Note that there were no ISFSI unloading/loading campaigns conducted in 2008.] In particular, the inspector reviewed licensee implementation of 0-SR-DCS3.1.2.1, Spent Fuel Storage Inspection, and 2-SR-

2, Table 1.41, Hi-Storm/Overpack Heat Removal System Operability. The inspectors also reviewed the special nuclear material (SNM) inventory forms of SPP-5.8, Special Nuclear Material Control, for the most recently loaded Hi-Storm cask transferred to the ISFSI pad (BFN-0-CASK-079-0100/5). Furthermore, the inspectors toured the ISFSI to verify configuration control of the loaded Hi-Storm casks in accordance with Certificate of Compliance (CoC) surveillance requirements. During this tour the inspectors also verified the locations of environmental dosimetry, examined radiological postings and radioactive material labels, and reviewed recent radiological dose rate and contamination surveys. In addition to routine operations activities, the inspectors also reviewed five 10 CFR 72.48 Screening Reviews for various ISFSI procedure and design changes, to verify these changes were consistent with the license and CoC, and did not reduce program effectiveness. [Note, none of the procedural or design changes conducted by the licensee since the last ISFSI inspection in 2007 required a full 10 CFR 72.48 evaluation, all changes were screened out by the screening reviews.]

b. Findings

No findings of significance were identified.

.3 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of Browns Ferry Nuclear Plant conducted in August 2008. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On January 9, 2008, the senior resident inspector presented the inspection results to Rusty West and other members of his 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 finding of very low safety significance (Green) was identified by the licensee and was a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- In March 2002, Unit 2 Reactor Zone Supply Secondary Containment Isolation Damper 2-DMP-64-14 failed in the open position due to the use of an improper lubricant in its solenoid valve. During an extent of condition review of this condition adverse to quality (i.e., defective material), the licensee determined that many of the solenoid valves used to control the secondary containment isolation dampers for Units 1, 2, and 3 were the same model, and susceptible to the same failure mode, as the solenoid valve for 2-DMP-64-14. According to 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that conditions adverse to quality, such as defective material and equipment, are to be promptly identified and corrected. However, contrary to Criterion XVI, the licensee failed to correct this adverse condition to quality on Unit 1 when they inadvertently omitted replacing the Unit 1 solenoid valves when the Unit 2 and 3 solenoid valves were replaced. The licensee identified this omission during surveillance testing in September 2008, when both of the Unit 1 Reactor Zone Exhaust Secondary Containment Isolation Dampers failed open (1-DMP-64-42 and 43) due to the improper lubricant in their solenoid valves. This finding was considered to be of very low safety significance because it only represented a degradation of the radiological barrier function of the reactor building secondary containment. The licensee entered this finding into the CAP as PER 152333.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

### KEY POINTS OF CONTACT

Licensee

S. Berry, Systems Engineering Manager
S. Bono, Director of Engineering
T. Brumfield, Training Manager
M. Cantrell, Operations Training Supervisor
D. Campbell, License Requalification Training Supervisor
P. Chadwell, Field Maintenance Superintendent
J. Davenport, Nuclear Licensing Engineer
S. Douglas, General Manager of Site Operations
A. Elms, Assistant General Manager Operations
J. Emens, Site Licensing Supervisor
D. Feldman, Interim Operations Manager
E. Frevold, Design Engineering Manager
K. Harvey, System Engineer
L. Hughes, Operations Superintendent
S. Ingram, ISFSI System Engineer
J. Johnson, Maintenance Performance Improvement/Work Control RCA Team Lead
M. Keck, Reactor Engineering Supervisor
D. Langley, Site Licensing Manager
J. LaCasse, System Engineer
F. Leonard, Inspection Services Organization
S. Lovvorn, Design Electrical Engineer/PM Program RCA Team Lead
S. Martin, Supervisor Mechanical Planning
G. McConnell, Component Engineer and PM Program Coordinator
J. Mitchell, Site Security Manager
M. Morrow, Operations Emergency Procedures
F. Nilsen, System Engineer
B. Quinn, Daily Scheduling Manager
E. Quinn, Performance Improvement Manager
D. Robertson, Operations Risk SRO
R. Rogers, Maintenance & Modifications Manager
P. Sawyer, Radiation Protection Manager
V. Schiavone, Mechanical/Nuclear Engineer
T. Shults, BOP Engineering Supervisor
H. Smith, Fire Protection Supervisor
R. Stowe, Nuclear Ops Support Superintendent
J. Underwood, Chemistry Manager
M. Welch, Inspection Services Organization
R. West, Site Vice President
J. Wolcott, Unit 1 Steam Dryer Modifications
J. Woodward, Equipment Reliability Manager

## LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000259,260,296/2008005-01 | NCV | Failure to Maintain Requalification Examination Integrity (Section 1R11.2) |

Closed

| | | |
|---|---|---|
| 05000259/2008-001-00 | LER | Loss of Safety Function Reactor Zone Exhaust Dampers Failed to Close (Section 4OA3.2) |

Discussed

None

## LIST OF DOCUMENTS REVIEWED

**Section 1R01: Adverse Weather Protection**

0-GOI-200-1, Freeze Protection Inspection, Rev. 60
PA-304, Freeze Protection Printout, dated 3/27/08
PA-304, Freeze Protection Printout, dated 12/17/08

**Section 1R04: Equipment Alignment**

Drawing 1-47E813-1, Flow Diagram Reactor Core Isolation Cooling System, Revision 29
1-OI-71, Attachment 1, Reactor Core Isolation Cooling System Valve Lineup Checklist, Rev. 3
1-OI-71, Attachment 2, Reactor Core Isolation Cooling System Panel Lineup Checklist, Rev. 4
1-OI-71, Attachment 3, Reactor Core Isolation Cooling System Electrical Lineup Checklist, Rev. 5
Technical Specifications and Bases 3.5.3, RCIC System, Amendment 269
2-47E812-1, Flow Diagram High Pressure Coolant Injection System, Rev. 54
2-OI-73 Attachment 1, High Pressure Coolant Injection Valve Lineup Checklist, 6/13/2008
2-OI-73 Attachment 2, High Pressure Coolant Injection Panel Lineup Checklist, 6/13/2008
2-OI-73 Attachment 3, High Pressure Coolant Injection Electrical Lineup Checklist, 3/29/2008
0-OI-82 Attachment 1A, Standby Diesel Generator A Valve Lineup Checklist, Rev. 98
0-OI-82 Attachment 2A, Standby Diesel Generator A Panel Lineup Checklist, Rev. 98
0-OI-82 Attachment 3A, Standby Diesel Generator A Electrical Lineup Checklist, Rev. 98
0-OI-82 Attachment 2, Standby Diesel Generator Common Panel Lineup Checklist, Rev. 98
0-OI-82 Attachment 3, Standby Diesel Generator Common Electrical Lineup Checklist, Rev. 98
0-OI-82 Attachment 1B, Standby Diesel Generator B Valve Lineup Checklist, Rev. 98
0-OI-82 Attachment 2B, Standby Diesel Generator B Panel Lineup Checklist, Rev. 98
0-OI-82 Attachment 3B, Standby Diesel Generator B Electrical Lineup Checklist, Rev. 98
0-47E840-3, Flow Diagram Fuel Oil System, Rev. 20
0-47E861-2, Flow & Control Diagram Diesel Starting Air System Diesel Generator B, Rev. 9
0-47E861-6, Flow Diagram – Cooling System and Lubricating Oil System Standby Diesel Generator B, Rev. 8

**Section 1R05: Fire Protection**

Fire Protection Report, Volume 1, Section I, Fire Protection Plan
Fire Protection Report, Volume 1, Section II, Fire Hazards Analysis, Rev. 1
Fire Protection Report, Volume 1, Section III, Safe Shutdown Analysis, Rev. 1
Fire Protection Report, Volume 2, Section IV.1, Pre-plan No RX1-519, Rev. 8
Fire Protection Report, Volume 2, Section IV.1, Pre-plan No RX1-519NW, Rev. 8
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No RX1-519SW, Rev. 7
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No RX1-565, Rev. 7
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No RX1-565, Rev. 8
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No RX1-519 SE, Rev. 7
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No RX1-519, Rev. 8
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No CB1-519 NE, Rev. 8
Fire Protection Report, Volume 2, Section IV.3, Pre-plan No RX1-593, Rev. 7
Fire Protection Report, Volume 2, Section IV.3, Pre-plan No RX1-621, Rev. 7
Fire Protection Report, Volume 2, Section IV.3, Pre-plan No RX1-639, Rev. 7

Fire Protection Report, Volume 2, Section IV.17, Pre-Plan TB3-586, Rev. 2
Active Fire Protection Impairments (FPIP'S)
0-SI-4.11.B.1.b, High Pressure Fire Protection Valve Position Verification (Inside Loop), Revision 47, 08/27/2008
0-SI-4.11.B.1.b, High Pressure Fire Protection Valve Position Verification (Inside Loop), Revision 47, 09/25/2008
0-SI-4.11.B.1.b, High Pressure Fire Protection Valve Position Verification (Inside Loop), Revision 47, 10/23/2008
1-SI-4.11.A.1(1), Local Fire Control Panel 1-LPNL-925-0545 Unit 1 Reactor Bldg Detection Operability Test, Revision 3, 1/15/2008
WO 08-711517-000, Reactor Building Fire Hose Shutoff Valve
WO 08-713498-000, Inspection of Portable and Wheel Type Fire Extinguisher Stations (Reactor Building) FP-0-000-INS001(A)
WO 08-714157-000, Inspection of Portable and Wheel Type Fire Extinguisher Stations (Reactor Building) FP-0-000-INS001(A)
WO 08-715278-000, Inspection of Portable and Wheel Type Fire Extinguisher Stations (Reactor Building) FP-0-000-INS001(A)
WO 08-718509-000, Reactor Building Battery and Battery Board Rooms Sprinkler Flow Valve
Unannounced or Announced Fire Drill Scenario, Rev 2., dated 10/21/2008

**Section 1R06: Flood Protection Measures**

UFSAR Section 10.9.3, RHRSW System, Rev. 22
GDC BFN-50-C-7105, Pipe Rupture, Internal Missiles, Internal Flooding, Seismic Equipment, Qualification and Vibration Qualification of Piping, Rev. 7
Browns Ferry Nuclear Plant Probabilistic Safety Assessment Internal Flooding Notebook, Rev. 1
Calculation MD-Q0023-870149, RHRSW Pump Compartment Sump and Sump Pump Capacity, Rev. 8
2-SI-3.3.13.C, ASME Section XI System Pressure Test of RHRSW Piping in the Intake Pits
(ASME Section III Class 3), Rev. 0 performed on 4/24/2008-4/25/2008
2-SI-3.3.14.C, ASME Section XI System Pressure Test of EECW Piping in the Intake Pits
(ASME Section III Class 3), Rev. 2 performed between 4/08/2008 and 4/23/2008
PER 135201, RHRSW/EECW Manhole Water
PER 135250, RHRSW/EECW Piping Below Ground Level at Intake
PER 136631, Water Leaking at Nitrogen Tank
PER 158456, Water Bubbling Up Out of the Ground in Front of RHRSW Pump Room
FE 42372, RHRSW/EECW Piping in Manhole
WO 08-724855-000, Repair/Grout the Vertical Joint Shown on 31N209 section B-B

**Section 1R08: Inservice Inspection Activities**

N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Rev. 32
N-UT-84, Procedure for the Phased Array Ultrasonic Examination of Austenitic and Ferritic Pipe Welds, Revision 0000GEIT-UT-1, Procedure for the Ultrasonic Examination of Austenitic and Ferritic Pipe Welds, Rev. 2
N-VT-1, Visual Examination Procedure for ASME Section XI Preservice and Inservice, Rev. 42
Problem Evaluation Report (PER) 127681, Work Order 06-716391-000 was closed without ASME Section XI Repair/Replacement and ANII review.
PER 141097, OE Unacceptable Flaw Found in RPV Cap to Pipe Weld

PER 142156, Prior to the 4/12/08 scheduled performance of 1-SI-3.2.31, errors were noted in Attachment 3
PER 145323, The Unit 1 ASME Section XI ISI Program 2nd Interval Update Has Not Been Submitted in a Timely Manner
PER 146647, INPO OE Digest 2008-02, Low Pressure Feedwater Heater Shell Leakage
Drawing 8029092D, Browns Ferry Unit 1 Steam Dryer Modification, Sheet 1 of 3, Rev. 4
Drawing 8029092D, Browns Ferry Unit 1 Steam Dryer Modification, Sheet 2 of 3, Rev. 4
Drawing 8029092D, Browns Ferry Unit 1 Steam Dryer Modification, Sheet 3 of 3, Rev. 4
Drawing 8029065C, Browns Ferry Unit 1 Steam Dam Gussett Fabrication, Sheet 1 of 2, Rev. 3
Drawing 8029065C, Browns Ferry Unit 1 Steam Dam Gussett Fabrication, Sheet 2 of 2, Rev. 3
Protocol GEIT-UT-1, PDI, Rev. 0

**Section 1R11: Licensed Operator Requalification Program**

Resident Inspector Quarterly Review
OPDP-1, Conduct of Operations, Rev. 9
Simulator Evaluation Guide for LOR-Exam-23, RPS Instrument Failure, Loss of RBCCW due to Leak Inside Drywell, Turbine Trip Without Bypass Valves, RPS Failure, Stuck Rods, LOCA, Failure of Drywell Spray Valves, Loss of Level Instrumentation, and RPV Flooding
Simulator Evaluation Guide for LOR-Exam-29, RWCU Isolation, MSL Pressure Transmitter Failure, Loss of RPS Bus, and ATWS with MSIV Closure
Simulator Evaluation Guide for LOR-Exam-42, Remove “B” High Pressure Heaters from Service, 2A RBCCW Pump Trip, Loss of 2B 480V S/D Board, Manual Scram, Steam Line Break in Containment, Spray PC Containment With Standby Coolant, and High Suppression level

Biennial Review
License Reactivation Packages (7)
Medical Files (17)
Remedial Training Records
Feedback Summaries
2007 Written Exam (07C6W3RO)
2007 Written Exam (07C6W3SRO)
TRN-12, Simulator Regulatory Requirements, Revision 8, 10/17/2008.
TRN-11.4, Continuing Training for Licensed Personnel, Revision 14, 07/25/2008.
TRN-11.10, Annual Requalification Examination Development and Implementation, Revision 13, 06/26/2008.
TRN-11.14, TVA Operator Licensing Examination Security Program, Revision 4, 09/01/2006.
TRN-11.12, Job Performance Measures Development, Administration, and Evaluation Manual, Revision 4, 07/31/2008.
TRN-11.11, Requalification Periodic Written Examination Development and Implementation, Revision 6, 09/25/2008.
Browns Ferry Unit 2, 100% Steady State Test, 12/13/2007.
Browns Ferry Unit 2, 75% Steady State Test, 12/13/2007.
Browns Ferry Unit 2, 50% Steady State Test, 12/13/2007.
Browns Ferry Unit 2, Stability Test (Drift), 12/13/2007.
Browns Ferry Unit 2 Transient Test, Tran 1 – Manual Scram, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 2 – Trip of All Feedwater Pumps, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 3 – Simultaneous Closure of All MSIVs, 11/29/2007.

Browns Ferry Unit 2 Transient Test, Tran 4 – Simultaneous Trip of All Recirculation Pumps, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 5 – Single Recirculation Pump Trip, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 6 – Main Turbine Trip No Scram, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 7 – Maximum Rate Power Ramp, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 8 – Maximum Size RPS Rupture With Loss of Offsite Power, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 9 – Maximum Size Unisolable Main Steam Line Rupture, 11/29/2007.
Browns Ferry Unit 2 Transient Test, Tran 10 – Closure of All MSIVs With Single Stuck Open SRV, 11/29/2007.
Browns Ferry Unit 2 Malfunction Test NM16, PRNM NUMAC Critical Failure, 12/13/2007.
Browns Ferry Unit 2 Malfunction Test FW19, Feedwater Line Break In The Main Steam Tunnel, 12/13/2007.
Browns Ferry Unit 2 Malfunction Test TH35, Main Steam Line Break in Steam Tunnel (Rx Bldg), 12/13/2007.
Browns Ferry Unit 2 Malfunction Test NM05, Variable Failure of IRM Attenuator Amplifier Output, 12/13/2007.
Browns Ferry Unit 2 Malfunction Test TC10, EHC Pressure Transducer Failure, 12/13/2007.
Browns Ferry Unit 2 Malfunction Test CS04, Core Spray Logic Power Failure, 12/13/2007.
Browns Ferry Unit 2 Malfunction Test ED27, 250V RMOV Board Breaker Failure, 12/13/2007.
Browns Ferry Unit 2 Malfunction Test RP06, Auto Scram Channels Fail (Manual Still Functional), 12/13/2007.
Browns Ferry Unit 2 Malfunction Test TH31, Reactor Pressure Inputs To EHC Control System, 12/13/2007.
Unit 3 Upper Power Runback / Core Flow Runback / Manual Scram (80% Power) 08/29/2006.
Unit 3 PLU Trip (100% Power) 12/31/2007.
Simulator Design Change Request B1477, Develop a Set of MOC ICs Consisting of HFP MELLA, HZP and COLD Conditions with U2C14 MOC Core Load Data From Nuclear Fuels Cycle Management Report, 05/18/2006.
Problem Report 4778, The DBM Database for the IC Globals Does Not Match Between Unit 2 and Unit 3, 04/12/2008.
Problem Report 4792, During LOR, the Simulator Tripped on High Water Level After a Single Recirc Pump Tripped from 95% Power Level, 06/12/2008.
Problem Report 4462, Tune Short-Term (Shrink) Level Response for Dual Recirc Pump Trip, and Long-Term (swell) Level Response for 40% Man Scram and 100% Turbine Trip, Based on U3 08/19/06 & 10/31/05 Transients, 09/06/06.
Problem Report 4104, The Response of the HW Level (2-3) and CST Makeup Flow (2-48) to Hotwell is not consistent with plant response. Makeup Flow does follow level, but response magnitude is excessive. Level Change is also different from the plant, 11/19/04.
LOR-Exam-32 Revision 1, Simulator Evaluation Guide, NI FAILURE, LOSS OF DG ‘D’, LOSS OF OFF-SITE POWER, HPCI STEAM LINE BREAK, LOCA, DIESEL AND CORE SPRAY FAILURES, May 9, 2008.
LOR-Exam-26 Revision 1, Simulator Evaluation Guide, LOSS OF SLC PUMP 2A, LOSS OF UPS, RECIRC PUMP VIBRATION/SEAL FAILURE, TURBINE TRIP, ATWS, FAILED OPEN MSRV, STEAM LINE BREAK ON RCIC, February 23, 2008.
LOR-Exam-33 Revision 1, Simulator Evaluation Guide, SJAE A ISOLATION, LOSS OF 4kv SHUTDOWN BOARD CONTROL POWER, SMALL LEAK IN DRYWELL, MAIN TURBINE TRIP W/O BYPASS VALVES, ATWS, LARGE BREAK LOCA, HPCI STEAM LEAK, May 10, 2008.

JPM 148, PERFORM IMMEDIATE OPERATOR ACTION FOR CRD PUMP TRIP, Revision 9, 06/27/2008.
JPM 14, 2-EOI APPENDIX 5A – INJECTION SYSTEMS LINEUP – CONDENSATE/FEEDWATER, Revision 11, 06/12/2008.
JPM 177TCF, CLASSIFY THE EVENT PER THE REP (ALERT – SECONDARY CONTAINMENT RADIATION), Revision 1, 05/29/2008.

**Section 1R12: Maintenance Effectiveness**

Maintenance Rule (a)(1) 10-Point Plan: U3 System 064A, Drywell Equipment Hatch (a)(1) Plan, Revision 0
PER 140438, U3C13 As-Found Administrative Leak Rates Exceeded, 03/19/2008
PER 98168, Unit 3 X-1A Equipment Hatch Failure, 03/01/2006
PER 95420, 3-SI-4.7.A.2.g-2/FH failed AC, 01/17/2006
PER 48404, As-Found Administrative leak rates exceeded during Unit 3 Cycle 11, 2/19/2004
PER 38911, As-Found Administrative leak rates exceeded during Unit 3 Cycle 10, 3/26/2002
Technical Specifications, 3.6.1.1 Primary Containment Amendment 212
Technical Requirements Manual, Section 3.6.5 Nitrogen Makeup to Containment, Rev. 0
PER 150500, Pinhole Leak in 2A/2C RHRSW Tunnel
PER 157264, RHRSW Piping 10 Point Plan
PER 49501, 2B RHRSW Piping Leak
PER 52847, U1 RHRSW Pipe Tunnel Water Leaks from Top of the Tunnel
PER 141734, Weeping Pinhole Leak in 2A RHRSW Inlet Header
Unit Common System 23, RHRSW/EECW System Health Report Card, FY2008 – P3
Unit Common RHRSW, Functions 23-B, C, and D (a)(1) Plan, Rev. 0
Cause Determination Evaluation (CDE) 680, Weeping Pinhole Leak in 2A RHRSW Inlet Header
General Design Criteria Document BFN-50-7023, Residual Heat Removal Service Water System, Rev. 12
FSAR Section 4.8, Residual Heat Removal System, BFN-22
Technical Specifications and Bases 3.7.1 RHRSW System and Ultimate Heat Sink, Amendment 254
Maintenance Rule (a)(1) 10-Point Plan: U2 RHR, Functions 074-C (a)(1) Plan, Rev. 0
CDE 544, 2-FCV-074-75 Failed to Open During Testing, dated 12/08/2006
CDE 666, 2-FCV-074-75 Failed to Open During Testing, dated 02/01/2008
PER 116191, 2-FCV-74-75 Failed to Open
PER 117948, Maintenance Rule Functional Failure
PER 140855, U2 MR Functional Failure
PER 149563, U2 RHR Loop II DW Spray BKR in MR (a)(1)

**Section 1R13: Maintenance Risk Assessments and Emergent Work Control**

BP-336, Risk Determination and Risk Management, Rev. 7
0-TI-367, BFN Equipment to Plant Risk Matrix, Rev. 10
Computer Sentinel Run for November 03, 2008
BFN-0-08-007, PRA Evaluation Response, dated 10/28/08
Operator Daily Logs
Unit 1 SFP Temperature Trend Charts
SPP-7.2.2, Outage Schedule Logic Change Form dated 11/21/08
ORAM Assessment for 11/20/08 and 11/21/08
WO 08-724192, Lubricate Bearings of 1A Fuel Pool Cooling Pump Motor

PER 1577821, 1B SFP Cooling Pump Trip and Safety Incident
BP-336, Risk Determination and Risk Management, for December 10 and 11.
Sentinel results for December 10 and 11, 2008

**Section 1R15: Operability Evaluations**

PER 134465, RHRSW Flow Rates
FE 42341, RHRSW Operability Issues Due to Degraded Flow Capabilities
Calculation MDQ002320030031, RHRSW System Flow Rates for RHR Heat Exchangers for Units 1, 2, and 3, 10/12/2006
Procedure 0-OI-23, RHR Service Water System, Rev. 87
Functional Evaluation 42922 for PER 154175

FSAR Appendix 2.4A, Maximum Possible Flood, BFN-22
FSAR Section 12.2.9, Equipment Access Lock (Class I), BFN-22
PER 151971, Door Seal Missing
PER 153527, Missing Flood Seals Beneath Equipment Access Lock Flood Gate
PER 154175, Past Operability for Missing Flood Seals
0-AOI-100-3, Flood Above Elevation 558', Rev. 33
MPI-0-000-INS001, Inspection of Flood Protection Devices, Rev. 10
WO 08-722061-000, Procurement and Staging of Flood Gate Seals
WO 08-722201-000, Preparation and Installation of Flood Gate Seals
DCN 63691, Modification to Equipment Access Lock/Roadway, Rev. A
Drawing 0-44N236-1, Reactor Building Equipment Access Lock Flood Gate Arrangement, Rev. 0
Drawing 44N236-2, Sheet 1, Reactor Building Equipment Access Lock Flood Gate Details, Rev. A
BFN-50-C-7100, Attachment C, Table 15-34, Equipment Access Lock Flood Gate Design Data, Rev. 18

PER 156416, DG Unit Priority Trip Relay Common Cause
PER 156401, DG C Unit Priority Retrip Relay Failures
0-TI-403, Determination of Common Cause Failure for Emergency Diesel Generators, Rev. 1, completed Appendices A and B dated 11/06/08
WO 08-723558-001 and -002 Technical Evaluations, Rev. 0
Drawing 0-45E765-11, Wiring Diagram Elementary Diagram, Emergency Equipment Wiring Schematic, Rev. 39
Drawing 0-731E761-14, Wiring Diagram, Elementary Diagram, Emergency Equipment Wiring Schematic, Rev. 28
Drawing 3-45E766-10, Wiring Diagram, 4160V Shutdown Aux Power, Schematic Diagram, Rev. 13
Drawing 3-730E938, Elementary Diagram, Residual Heat Removal System, Revision 10
FSAR Section 8.5.4.2, Diesel Generator Loading
Technical Specification and Bases 3.8.1, A.C. Sources – Operating, Amendment 249

PER 157270, Welding Rod dropped in equipment pit during Dryer repairs
Unit 1 - Lost Welding Rod Evaluation, TVA-BFNP Reactor Engineering, Rev. 1 dated 11/26/2008
Unit 1 - List of Non-Recovered Items, TVA-BFNP Reactor Engineering, Dated 12/15/2008

PER 155534, Instrument Malfunction Causes U1 Water Level Transient during Cooldown
FE 42956, U1 Water Level Transient during Cooldown
PER 14993,1 Reactor Water Level Perturbation

PER 156982, U1C7 Jet Pump Restrainer Bracket Indications
PER 156982,Unit 1 Jet Pump Restrainer Bracket Adjusting Set Screw Tack Weld Cracks
General Electric (GE) SIL 574, Jet Pump Adjusting Screw Tack Weld Failures
WO 08-725211, Contingency To Perform Additional Visual Exams of Jet Pumps #11 and 16
GE – Hitachi (GEH) 0000-0093-7911-R0, November 2008, Engineering Report of the Jet Pump Visual Inspection of Browns Ferry Nuclear Plant RF07
Structural Integrity Associates Report No. 0801464.401.R0, November 18, 2008, BFN Unit 1 Cracked Jet Pump Set Screw Tack Welds Evaluation
TVA Record #R06 081127 901, November 27, 2008, Review of Cycle 7 Refueling Outage Reactor Jet Pump Restrainer Bracket Assembly Inspection Results
BWR Vessel and Internals Project (BWRVIP)-41, BWR Jet Pump Assembly Inspection and Flaw Evaluation

**Section 1R19: Post-Maintenance Testing**

1-SR-3.3.5.1.6(A II), Functional Testing of RHR Loop II Automatic Initiation Logic and Injection Valve Opening Pressure Permissive Logic, Rev. 6
2-SR-3.3.5.1.6(A II), Functional Testing of RHR Loop II Automatic Initiation Logic and Injection Valve Opening Pressure Permissive Logic, Rev. 12
WO 08-723558-001, Replacement of C EDG Output Breaker 1812 Unit 1 Priority Retrip Relay
WO 08-723558-002, Replacement of C EDG Output Breaker 1812 Unit 2 Priority Retrip Relay
ECI 0-000-RLY003, Replacement of Relays, Rev. 20
1-SR-3.5.3.3(COMP), RCIC Comprehensive Pump Test, Rev. 8, Performed 12/15/2008
ECI-0-071-GOV001, RCIC Governor Control System Calibration, Rev. 0028, Performed 12/12/2008, 12/04/2008, 12/01/2008
PER 159148, Procedure steps signed in error
PER 159257, RCIC Inoperable
PER 159575, Performance of Cold Quick Start Testing on HPCI and RCIC
WO 07-726220-000, EGR Hydraulic Actuator – Replace the Existing Unit-1 RCIC Actuator with a Refurbished Model. Return the Existing/Old EG-R to the Manufacturer for Rebuild
WO 08-714794-000, RCIC Turbine Governor – Perform Pre-Dynamic/Dynamic Inspection and Calibration of the RCIC Turbine Governor Control System per ECI-0-071-GOV001
WO 08-725178-000, RCIC Turbine Ramp / Gen Signal Converter – Support RCIC Troubleshooting, Static and/or Dynamic Tuning/Testing at the Direction of the Cognizant Engineer
1-SR-3.5.1.6 (RHR II COMP) RHR Loop II Comprehensive Pump Test, Rev. 7

**Section 1R20: Refueling and Other Outage Activities**

1-TO-2008-003, Clearance 1-074-0004A for 1-CKV-074-0559A and RHR Pump 1A Discharge Check Valve
1-TO-2008-003, Clearance 1-023-002 for RHR HX 1D (RHRSW Side)
SPP-10.2 Clearance Procedure to Safely Control Energy, Rev. 12

**Section 1R22: Surveillance Testing**

Technical Specifications Section 3.3.3.2, Backup Control System
BFN UFSAR Section 7.18, Backup Control System
1-SR-3.3.3.2.1(75 II), Backup Control Panel Testing, Rev. 1
Technical Specifications Section 3.6.1.3, Primary Containment Isolation Valves
Drawing 2-47E860-1, Flow Diagram Containment Inerting System
BFN UFSAR Section 5.2, Primary Containment System
2-SR-3.6.1.3.5(76 II), H2O2 System Isolation Valve Operability Test (Division II), Rev. 14
Technical Specifications and Bases Section 3.5.1, ECCS - Operating
BFN UFSAR Section 6.4.1, High Pressure Coolant Injection System Description
3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure, Rev. 47
Drawing 3-47E812-1, Flow Diagram High Pressure Coolant Injection System, Rev. 59
Unit 1 Technical Specifications Bases, Rev. 58
1-SR-3.7.5.1, Turbine Bypass Valve Cycling, Rev. 1
1-SR-3.8.1.9(A), Diesel Generator A Emergency Unit 1 Load Acceptance Test, Rev. 4
PER 157838, U1/2 DG A LAT – Recorder B – All Data Not Captured
PER 157802, Surveillance Criteria Not Met
PER 157756, Breaker 1612 Found Open During SR When Previously Signed for as Closed
PER 157803, A DG Load Acceptance Test DAQ Failure
WO 08-714830, 0-TI-298 Diesel Generator Operating Data Acquisition, Rev. 12
Technical Specifications and Bases Section 3.4.3.2
1-SR-3.4.3.2, Main Steam Relief Valves Manual Cycle Test

**Section 1EP6: Drill Evaluation**

Emergency Plan Implementation Procedure (EPIP) 1, Emergency Classification Procedure, Rev. 43
EPIP 2, Notification of Unusual Event, Rev. 29
EPIP 3, Alert, Rev. 32
EPIP 4, Site Area Emergency, Rev. 31
Performance Indicator Data, 2008 BFN Green Team, 10/01/2008

**Section 4OA1: Performance Indicator Verification**

PER 160141, Value entered in CDE for Unit 2 failed to include an unplanned downpower in September of 2008

**Section 4OA2: Identification and Resolution of Problems**

Semi-Annual Trend Review
BP-336, Risk Determination and Risk Management, Revision 7
PER 150254, ITR Not Completed and Site Summary Not Issued
PER 151684, ITR Requirements Not Being Met
PER 157492, Deferment of ITR to 2009
PIDP-11, PER Trending, Revision 0
Departmental Integrated Trend Analysis Reports (ITR) for January to April 2008
Corrective Action Program Quality Index for January to April 2008 and November 2008
NPG Fleet CAP Health Monitor, November, 2008

PER 152007, Maintenance Rule Program Concerns
PER 126875, Maintenance Rule Focus
System Engineer Incomplete CDE Status report dated August 19, 2008
BFN Maintenance Rule SSCs in (a)(1) Status reports
System Engineer unreliability and unavailability MR system data

Focused Annual Sample Review – Preventive Maintenance (PM) Program
PER 153450, Root Cause Analysis Report TVA Browns Ferry Preventive Maintenance Program
PER 148701, Preventive Maintenance Program Issues (Turnaround) Plan
PER 152795, PM Program Not Maintaining Safe/Reliable Equipment Performance and Reliability
PER 152846, Necessary PM Tasks Not Established
PER 116524, PMs Not Implemented Per Schedule
PER 113976, PM Deferral Categories
PER 113974, PM Deferral Inadequate Technical Justifications
PER 111218, PM Deferral Benchmarking
PER 153435, Work Schedule Adherence
Turnaround Plan to Excellence Browns Ferry Nuclear Plant, Equipment Reliability, Preventive Maintenance Program Issues, Revision 4
PIDP-6, Root Cause Analysis, Revision 1
SPP-6.2, Preventive Maintenance, Revision 5
Outage/Scheduling Performance Indicator, Preventive Maintenance Grace Utilization, 12/30/08
Maintenance Planning Performance Indicator, PMs in Grace Period, 12/12/08 and 12/30/08

**Section 4OA5: Other Activities**

PER 129569, Refuel Floor Paint Damage from HI-TRAC
PER 133455, Delay in ISFSI Loading
SPP-5.8, Special Nuclear Material Control, Rev. 11
0-SR-DCS3.1.2.1, Spent Fuel Storage Inspection, Rev. 4
2-SR-2, Instruments Checks and Observations, Table 1.41, Rev. 61
MSI-0-079-DCS012, MPC Processing, Rev. 16
MSI-0-079-DCS008, Movement and Transfer Operations of HI-TRAC and HI-STORM in the Reactor Building, Rev. 9
MSI-0-000-LFT001, Lifting Instructions for Control of Heavy Loads, Rev. 48
Final Safety Analysis Report for the Holtec HI-STORM 100 Cask System, Rev. 2
Certificate of Compliance for Spent Fuel Storage Casks for Holtec HI-STORM 100 Cask System, Docket 72- 014, Amendment 1, including Appendix A (Technical Specifications), Appendix B (Approved Contents and Design Features)
DCN63689, Units 1 and 3 Spent Fuel Pool HI-TRAC Base Plate Modification, Revision A
10 CFR 72.48 Screening, MSI-0-000-LFT001 Revision 47C, Rev. 0
10 CFR 72.48 Screening, MSI-0-000-LFT001 Revision 48, Rev. 0
10 CFR 72.48 Screening, DCN 63689A, Rev. 0
10 CFR 72.48 Screening, MSI-0-079-DCS008 Revision 9, Rev. 0
10 CFR 72.48 Screening, MSI-0-079- DCS012 Revision 16, Rev. 0

## LIST OF ACRONYMS

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|---|---|
| ADAMS | Agencywide Document Access and Management System |
| ADHR | Auxiliary Decay Heat Removal |
| CAP | corrective action program |
| CFR | *Code of Federal Regulations* |
| CoC | certificate of compliance |
| CRD | control rod drive |
| DCN | design change notice |
| EECW | emergency equipment cooling water |
| EDG | emergency diesel generator |
| FSAR | Final Safety Analysis Report |
| IMC | Inspection Manual Chapter |
| JPM | Job Performance Measure |
| LER | licensee event report |
| NCV | non-cited violation |
| NRC | U.S. Nuclear Regulatory Commission |
| PER | problem evaluation report |
| PI | performance indicator |
| RG | Regulatory Guide |
| RHR | residual heat removal |
| RHRSW | residual heat removal service water |
| RTP | rated thermal power |
| RPS | reactor protection system |
| SDP | significance determination process |
| SLC | standby liquid control |
| SNM | special nuclear material |
| SRV | safety relief valve |
| SSC | structure, system, or component |
| TI | Temporary Instruction |
| TS | Technical Specification(s) |
| UFSAR | Updated Final Safety Analysis Report |
| WO | work order |