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

May 14, 2013

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2013002, 05000260/2013002, AND 05000296/2013002, AND ASSESSMENT FOLLOW-UP

Dear Mr. Shea:

On March 31, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. The enclosed inspection report documents the inspection results which were discussed on April 5, and 26, 2013, with Mr. Keith Polson 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, orders, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The enclosed inspection report discusses two findings that have the potential to be of greater than very low safety significance (Green) resulting in the need for further evaluation to determine significance and need for additional NRC action. The first finding is discussed in Section 1R15 of the enclosed report and is associated with the failure to establish an adequate preventive maintenance program to maintain the Residual Heat Removal Service Water Pump D1 Cross-Tie to Emergency Equipment Cooling Water Valve (0-FCV-067-0048), in a manner that ensured it would perform its design function. Although this finding has potential safety significance, it does not present a current safety concern because the licensee has repaired the valve. The second finding is discussed in Section 4OA3.3 of the enclosed report and is associated with the failure to correctly follow procedures associated with restoration of 2B Reactor Protection System (RPS) 480 volt power. Although this finding has potential safety significance, it does not present a current safety concern because the licensee has taken actions to prevent recurrence of the associated human performance error that caused the event. These findings are being assessed based on the best available information, using the applicable Significance Determination Process. The final resolution of these findings will be conveyed in separate correspondence.

These findings are also apparent violations of NRC requirements and are being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

In accordance with NRC Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation, using the best available information, and issue our final determination of safety significance within 90 days of the date of this letter. The significance determination process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Additionally, two licensee-identified violations which were determined to be of very low safety significance (Green) are listed in this report. The NRC is treating the violations as non-cited violations (NCV) consistent with Section 2.3.2 of the Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to: (1) the Regional Administrator, Region II; (2) the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and (3) the NRC Resident Inspector at the Browns Ferry Nuclear Plant.

If you disagree with any cross-cutting aspect assignment in the report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Browns Ferry Nuclear Plant.

As a result of its quarterly review of plant performance, which was completed on April 30, 2013, the NRC updated its assessment of Browns Ferry Nuclear Plant Unit 2. The NRC's evaluation consisted of a review of performance indicators and inspection results. The NRC's review of Browns Ferry Nuclear Plant Unit 2 identified that the Emergency AC Power Systems performance indicator in the Mitigating Systems Cornerstone has crossed the Green-to-White threshold beginning the fourth quarter 2012. In combination with the White finding documented in inspection report 05000259, 260, 296/2012013 (ADAMS Accession Number ML12226A647), also in the Mitigating Systems Cornerstone, the NRC has assessed the performance of Browns Ferry Nuclear Plant Unit 2 to be in the Degraded Cornerstone column of the Reactor Oversight Process Action Matrix beginning the fourth quarter of 2012. This letter supplements, but does not supersede, the annual assessment letter issued on March 4, 2013 (ADAMS Accession Number ML13063A461).

The NRC will conduct a supplemental inspection (Inspection Procedure 95002) when you have notified us of your readiness for the NRC to review the actions taken to address each of these issues. This inspection will review both the White inspection finding and White performance indicator. This inspection procedure is conducted to provide assurance that the root and contributing causes for the individual and collective risk significant performance issues are understood, to independently assess the extent of condition, to provide assurance that the corrective actions are sufficient to prevent recurrence, and to independently determine if safety culture components caused or significantly contributed to individual and collective risk-significant performance issues.

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.

Sincerely,

/RA/

Victor M. McCree
Regional Administrator

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

Enclosure: NRC Integrated Inspection Report 05000259/2013002, 05000260/2013002, and 05000296/2013002

cc w/encl: (See page 4)

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.

Sincerely,

/RA/

Victor M. McCree
Regional Administrator

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

Enclosure: NRC Integrated Inspection Report 05000259/2013002, 05000260/2013002, and 05000296/2013002

cc w/encl: (See page 4)

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

ADAMS: ☐ Yes ACCESSION NUMBER:________________________ X SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRP
SIGNATURE	Via email	Via email	Via email	Via email	Via email	Via email	DXW /RA/
NAME	DDumbacher	CStancil	PNiebaum	LPressley	TStephen	CFletcher	DJackson
DATE	05/02/2013	05/02/2013	05/02/2013	05/02/2013	05/02/2013	05/06/2013	05/03/2013
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:ORA	RII:ORA		
SIGNATURE	CRK /RA/		RPC /RA/	LDW /RA/	VMM /RA/		
NAME	CKontz	WJones	RCroteau	LWert	VMcCree		
DATE	05/06/2013		05/08/2013	05/13/2013	05/14/2013		
E-MAIL COPY?							

OFFICIAL RECORD COPY DOCUMENT NAME: G:\DRPII\RPB6\BROWNS FERRY\REPORTS\2013\002\BROWNS FERRY 2013002.DOCX

cc w/encl :
K. J. Polson
Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

S. M. Bono
Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

James E. Emens
Manager, Site Licensing
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

E. W. Cobey
Manager, Corporate Licensing
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

T. A. Hess
Program Manager
Corporate Licensing
Tennessee Valley Authority
Electronic Mail Distribution

Edward J. Vigluicci
Associate General Counsel, Nuclear
Tennessee Valley Authority
Electronic Mail Distribution

Chairman
Limestone County Commission
310 West Washington Street
Athens, AL 35611

State Health Officer
Alabama Dept. of Public Health
P.O. Box 303017
Montgomery, AL 36130-3017

Letter to Joseph W. Shea from Victor M. McCree dated May 14, 2013

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2013002, 05000260/2013002, AND 05000296/2013002, AND ASSESSMENT FOLLOW-UP

DISTRIBUTION:
C. Evans, RII
L. Douglas, RII
OE Mail
RIDSNRRDIRS
PUBLIC
RidsNrrPMBrownsFerry Resource

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.:	05000259/2013002, 05000260/2013002, 05000296/2013002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	January 1, 2013, through March 31, 2013
Inspectors:	D. Dumbacher, Senior Resident Inspector C. Stancil, Resident Inspector P. Niebaum, Resident Inspector L. Pressley, Resident Inspector T. Stephen, Resident Inspector C. Fletcher, Senior Reactor Inspector
Approved by:	Craig Kontz, Acting Chief Special Project Browns Ferry Division of Reactor Projects

Enclosure

SUMMARY

IR 05000259/2013002, 05000260/2013002, 05000296/2013002; 01/01/2013–03/31/2013; Browns Ferry Nuclear Plant, Units 1, 2, and 3; Operability Evaluations and Follow-up of Events.

The report covered a three month period of inspection by the resident inspectors and one regional inspector. Two self-revealing findings were identified. Additionally, two licensee-identified violations of very low safety significance (Green) are documented. The significance of most findings is identified by their color (Green, White, Yellow, and Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); and, the cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." 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.

List of Findings and Violations

Cornerstone: Initiating Events

- TBD. A self-revealing Apparent Violation (AV) of Technical Specification 5.4.1 was identified for the licensee's failure to properly implement procedure 2-OI-99, Reactor Protection System. Specifically, during restoration of 2B Reactor Protection System (RPS) 480 volt power, the RPS motor generator set tie to battery BD 2 Breaker on the 2A RPS bus motor generator set was incorrectly opened. The licensee took immediate actions to respond to the resultant Unit 2 scram and placed the unit in a shutdown condition. Subsequent corrective actions included operator training and procedure revisions. The licensee entered this issue into their corrective action program as Problem Evaluation Report (PER) 660862.

  This finding was determined to be more than minor because it was associated with the Initiating Events cornerstone attribute of the human performance area and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. Specifically, the failure to properly implement procedure 2-OI-99 caused a Unit 2 reactor scram and main steam isolation valves (MSIV) closure. Because the finding could not be screened as very low safety significance (Green), nor its safety significance determined prior to issuing the inspection report, it is being characterized as "To Be Determined (TBD)." The cause of this finding was directly related to the cross-cutting aspect of Human Error Prevention in the Work Practices component of the Human Performance area, because the lack of adequate self-check, peer checking, and pre-job briefing resulted in the operator opening the incorrect breaker. [H.4(a)], (Section 4OA3.3)

Cornerstone: Mitigating Systems

- TBD. A self-revealing Apparent Violation (AV) of 10 CFR 50 Appendix B, Criterion V, Instructions, Procedures, and Drawings, was identified for the licensee's failure to establish an adequate preventive maintenance program as required by procedure NPG-SPP-06.2, Preventive Maintenance. Specifically, the Residual Heat Removal Service Water Pump D1 Cross-Tie to Emergency Equipment Cooling Water Valve (0-FCV-067-0048), was not maintained in a manner that ensured it would perform its design function. The failed valve was replaced on January 16, 2013, with a new valve with a stainless steel disk. Further corrective actions were planned to develop adequate preventive maintenance activities for this valve. The licensee entered this issue into their corrective action program as PER 671314.

  This finding was determined to be more than minor because it was associated with the Protection Against External Events (fires) attribute of the Mitigating Systems cornerstone objective and adversely affected the cornerstone objective to ensure availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the 0-FCV-067-0048 valve failed and could not perform its isolation function credited in the safe shutdown analysis. Because the finding could not be screened as very low safety significance (Green), nor its safety significance determined prior to issuing the inspection report, it is being characterized as "To Be Determined (TBD)." The cause of this finding was directly related to the cross-cutting aspect of Appropriately Coordinating Work Activities in the Work Control component of the Human Performance area, because maintenance activities for 0-FCV-067-0048 were more reactive than preventive. [H.3(b)], (Section 1R15)

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

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full Rated Thermal Power (RTP) except for four planned downpowers, one unplanned downpower, and one manual reactor scram. On January 11, 2013, a planned downpower to 50 percent was performed for control rod pattern adjustments. On January 13, 2013, an unplanned downpower to 93 percent was performed to repair a steam leak on 1B feedwater pump. On March 8, 2013, a planned downpower to 68 percent was made to perform control rod pattern adjustments, scram time testing, and main turbine valve testing. On March 15, 2013, a planned downpower to 95 percent was made to perform 1A feedwater pump relay power supply replacements. On March 19, 2013, Unit 1 was manually scrammed due to failure of 1C2 feedwater heater tubes which caused a loss of vacuum in the main condenser. On March 30, 2013, a planned downpower to 75 percent was made to perform feedwater heater room inspections, and returned to full power the same day. The unit remained at full power the remainder of the quarter

Unit 2 operated at full RTP except for one planned downpower and one planned scram to initiate refueling outage U2R17. On January 18, 2013, a planned downpower to 55 percent was performed for main steam dump valve repairs and final feedwater temperature reduction to maximize power. On March 14, 2013, Unit 2 was manually scrammed to enter U2R17 early. Original outage start was planned for March 17, 2013. The early start was due to anticipation that RCIC Technical Specification LCO 3.5.3 Action Statement B would not be completed on time. The unit remained in a shutdown condition the remainder of the quarter.

Unit 3 operated at full RTP power except for one planned downpower and one unplanned automatic reactor scram. On February 11, 2013, Unit 3 shutdown for planned repairs of circulating water condenser supply underground piping, 3A recirculation pump suction valve motor and miscellaneous drain header. After the repairs the unit was returning back to full RTP when an automatic reactor scram, caused by a loss of vacuum in the main condenser, occurred on February 25, 2013. The unit returned to full power on February 28, 2013, and remained at full power the remainder of the quarter.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 External Flood Protection

a. Inspection Scope

The inspectors reviewed plant design features and licensee procedures intended to protect the plant and its safety-related equipment from external flooding events. The inspectors reviewed flood analysis documents including: Updated Final Safety Analysis Report (UFSAR) Section 2.4, Hydrology, Water Quality, and Marine Biology, which included Appendix 2.4A, Maximum Possible Flood. The inspectors performed walkdowns of risk-significant areas which contained susceptible systems and equipment. Specifically the inspectors reviewed select flood barriers associated with the residual heat removal service water (RHRSW) system power conduit penetrations into the RHRSW pump rooms from the intake pump station. Evaluations concerning these penetrations were contained in Problem Evaluation Reports (PERs) 666222 and 671475.

b. Findings

No findings 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, listed below, while the other train or subsystem was inoperable or out of service. The inspectors reviewed the functional systems descriptions, 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. This activity constituted three Equipment Alignment Partial Walkdown inspection samples.

- Temporary tank release path for the Radiological Waste Building
- Unit 3 High Pressure Coolant Injection (HPCI) System
- Units 1, 2, and 3 RHRSW System

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Units 1 and 2, B EDG, using the applicable diagrams, 0-47E861-2A, 0-47E861-5, and 0-47E840-3, along with the relevant operating instructions, 0-OI-18 and 0-OI-82, to verify equipment availability and operability. The inspectors reviewed relevant portions of the UFSAR and TS. This detailed walkdown also verified electrical power alignment, the condition of applicable system instrumentation and controls, component labeling, pipe hangers and support installation, and associated support systems status. Furthermore, the inspectors examined applicable System Health Reports, open work orders (WOs), and any previous PERs that could affect system alignment and operability.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Fire Protection Tours

a. Inspection Scope

The inspectors reviewed licensee procedures, Nuclear Power Group Standard Programs and Processes (NPG-SPP)-18.4.7, Control of Transient Combustibles, and NPG-SPP-18.4.6, Control of Fire Protection Impairments, and conducted a walkdown of five fire areas (FA) and fire zones (FZ) listed below. Selected FAs and FZs were examined 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 NPG-SPP-18.4.6. Furthermore, the inspectors reviewed applicable portions of the Fire Protection Report, Volumes 1 and 2, including the applicable Fire Hazards Analysis, and Pre-Fire Plan drawings, to verify that the necessary firefighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, was in place. This activity constituted five inspection samples.

- Fire Area 15, Unit 3 Reactor Building, EL 621’, 480V Shutdown Board Room 3B
- Fire Area 19, Unit 3 Control Building, EL 593’, Battery and Battery Board Room
- Fire Area 20, Unit 1 and 2, EL 565’ and 583’, Emergency Diesel Generator Building

- Fire Area 21, Unit 3 Diesel Generator Building, All Elevations excluding 4kV Shutdown Board Rooms
- Fire Area 22, Unit 3, EL 565.5 feet and EL 583.5 feet, 4 KV Shutdown Board Room 3EA and 3EB, Unit 3 Diesel Generator Building

b. Findings

No findings were identified.

.2 Annual Fire Brigade Drill:

a. Inspection Scope

On March 13, 2013, the inspectors witnessed an unannounced fire drill in the Unit 1 Turbine Building (EL 565 feet) at the Condensate Transfer Pump ‘A’ Motor.

b. Findings

No findings were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed a review of the Units 1 and 2 Emergency Diesel Generator (EDG) rooms for internal flood protection measures. 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 and the licensee’s analysis for internal and external flooding events effects on the EDG rooms.

The inspectors performed a walkdown of risk-significant areas, susceptible systems and equipment, including the Units 1 and 2 EDG rooms to review flood-significant features such as area level switches, room sumps and sump pumps, flood protection door seals, conduit seals and instrument racks that might be subjected to flood conditions. Plant procedures for mitigating flooding events were also reviewed to verify that licensee actions were consistent with the plant’s design basis assumptions.

The inspectors also reviewed a sample 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, and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08G, Unit 2)

a. Inspection Scope

Non-Destructive Examination (NDE) Activities and Welding Activities: From March 25 to March 28, 2012, the inspectors conducted an on-site review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, emergency feedwater systems, risk-significant piping and components, and containment systems in Unit 2. The inspector's activities included a review of non-destructive examinations (NDEs) to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2004 Edition with no Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI, acceptance standards.

The inspectors directly observed the following NDE mandated by the ASME Code to evaluate compliance with the ASME Code Section XI and Section V requirements and, if any indications and defects were detected, to evaluate if they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- UT Exam of Weld SLCS N-10-1 (Standby Liquid Control System), Category B-J
- UT (Phased Array) Exam of Weld HPCISTHPCI-2-070 (High Pressure Coolant Injection System), Category R-A
- UT (Phased Array) Exam of Weld CSS DCS-2-13 (Core Spray System)

The inspectors reviewed records of the following NDEs mandated by the ASME Code Section XI to evaluate compliance with the ASME Code Section XI and Section V requirements and, if any indications and defects were detected, to evaluate if they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- VT-3 of 2-47B455H0066, High Pressure Coolant Injection System Variable Support
- Liquid Penetrant Exam for Component RHR-2-037-033 COR0, Residual Heat Removal System

The inspectors reviewed associated documents for the welding activities referenced below to evaluate compliance with procedures and the ASME Code. The inspectors reviewed the work order, repair and replacement plan, weld data sheets, welding procedures, procedure qualification records, welder performance qualification records, and NDE reports.

- Welding Package for Component RHR-2-037-034-COR0

During non-destructive surface and volumetric examinations performed since the previous refueling outage, the licensee did not identify any relevant indications that were analytically evaluated and accepted for continued service. Therefore, no NRC review was completed for this inspection procedure attribute.

Identification and Resolution of Problems: The inspectors performed a sample review of ISI-related problems which were identified by the licensee and entered into the corrective action program as PERs. The inspectors reviewed the PERs to confirm the licensee had appropriately described the scope of the problem, and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On January 22, 2013, the inspectors observed an as-found licensed operator requalification simulator examination for an operating crew according to Unit 2 Simulator Exercise Guide OPL177.083, Manual Scram, Steam Line Break in Containment, Use of Containment Spray with Standby Coolant System.

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 procedures including Abnormal Operating Instructions (AOIs), and Emergency Operating Instructions (EOIs)
- Timely control board operation and manipulation, including high-risk operator actions
- Timely oversight and direction provided by the shift supervisor, including ability to identify and implement appropriate technical specifications actions such as reporting and emergency plan actions and notifications
- Group dynamics involved in crew performance

The inspectors assessed the licensee's ability to administer testing and assess the performance of their licensed operators. The inspectors attended the post-examination critique performed by the licensee evaluators and verified that licensee-identified issues were comparable to issues identified by the inspector. The inspectors also reviewed simulator physical fidelity (i.e., the degree of similarity between the simulator and the reference plant control room, such as physical location of panels, equipment, instruments, controls, labels, and related form and function). This activity constitutes one inspection sample.

b. Findings

No findings were identified.

.2 Control Room Observations

a. Inspection Scope

Inspectors observed and assessed licensed operator performance in the plant and main control room, particularly during periods of heightened activity or risk and where the activities could affect plant safety. Inspectors reviewed various licensee policies and procedures such as OPDP-1, Conduct of Operations, NPG-SPP-10.0, Plant Operations and GOI-100-12, Power Maneuvering.

Inspectors utilized activities such as post maintenance testing, surveillance testing and refueling and other outage activities to focus on the following conduct of operations as appropriate:

- Operator compliance and use of procedures.
- Control board manipulations.
- Communication between crew members.
- Use and interpretation of plant instruments, indications and alarms.
- Use of human error prevention techniques.
- Documentation of activities, including initials and sign-offs in procedures.
- Supervision of activities, including risk and reactivity management.
- Pre-job briefs.

This activity constituted one Control Room Observation inspection sample.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

.1 Routine

a. Inspection Scope

The inspectors reviewed the specific structures, systems and components (SSC) within the scope of the Maintenance Rule (MR) (10 CFR 50.65) with regard to some or all of the following attributes, as applicable: (1) Appropriate work practices; (2) Identifying and addressing common cause failures; (3) Scoping in accordance with 10 CFR 50.65(b) of the MR; (4) Characterizing reliability issues for performance monitoring; (5) Tracking unavailability for performance monitoring; (6) Balancing reliability and unavailability; (7) Trending key parameters for condition monitoring; (8) System classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); (9) Appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2); and (10) Appropriateness and adequacy of 10 CFR 50.65 (a)(1) goals, monitoring and corrective actions (i.e., Ten Point Plan). The inspectors also compared the licensee's performance against site procedure NPG-SPP-3.4, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; and NPG-SPP 3.1, Corrective Action Program. The inspectors also reviewed, as 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. This activity constituted one Maintenance Effectiveness inspection sample.

- Units 1, 2, and 3 Rod Worth Minimizer (RWM)

b. Findings

No findings 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 examined five on-line maintenance risk assessments, and actions taken to plan and/or 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) applicable plant procedures, and BFN Equipment to Plant Risk Matrix. Furthermore, as applicable, the inspectors verified the actual in-plant configurations to ensure accuracy of the licensee's risk assessments and adequacy of RMA implementations. This activity constituted five Maintenance Risk Assessment inspection samples.

- January 29, 2013; 3A EDG and Unit 3 RCIC out of service with the potential for general thunderstorms and wind advisory
- February 28, 2013; Unit 3 Application of TS LCO 3.0.4.b, Mode Change Risk Assessment, Due to Isolation of the Entire A RHRSW Header
- March 4, 2013; Unit 2 Reactor Core Isolation Cooling (RCIC) Emergent Turbine Exhaust Valve Failure with A Emergency Diesel Generator (EDG), A3 Emergency Equipment Cooling Water (EECW) pump and strainer, Service Air Compressors, and Raw Cooling Water (RCW) Pumps 1D and 3D Out of Service (OOS)
- March 15, 2013; Evaluation of initial Unit 2 Outage risk and impact and effect upon operating risk of Unit 1
- March 26-29, 2013; Yellow Outage Risk Unit 2 for 24 Control Rod Drive exchanges (Operation with Potential to Drain the Reactor Vessel, OPDRV)

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed Technical Specification operability. The inspectors also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedure NEDP-22, Functional Evaluations, and NEDP-27, Past Operability 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. This activity constituted nine Operability Evaluation inspection samples.

- Units 2 and 3 Residual Heat Removal (RHR) pump and Core Spray (CS) pump motor lead wire environmental qualification (PER 652786)
- Unit 2 Torus Dynamic Restraint (Snubber BFN-2-SNUB-64-11) low fluid level (PER 667765)
- RHRSW Pump Room A Potential Flooding Issues (PER 666222)
- Unit 3 RCIC (Reactor Core Isolation Cooling) operated below recommend RPM (PER 687912)
- Ground water leaks surrounding the Intake pump house area (PER 666225)
- Emergency diesel A manually tripped due to excessive loading during paralleling operation with 3A emergency diesel (PER 672780)
- Emergency diesels B, 3A, and 3B declared inoperable related to discovery of generator fan bearings lack of lubrication (PER 675339)
- Failure of the 0-FCV-067-0048 Residual Heat Removal Service Water Cross-Tie Valve (PER 671314)
- 3D Emergency Diesel Generator Blower Inboard Bearing Failure Past Operability (PER 665217)

b. Findings

Introduction: A self revealing apparent violation (AV) of 10 CFR 50 Appendix B, Criterion V, Instruction, Procedures, and Drawings, was identified for the licensee's failure to establish an adequate preventive maintenance program to maintain the 0-FCV-067-0048, Residual Heat Removal Service Water Pump D1 Cross-Tie to Emergency Equipment Cooling Water Valve, in a manner that ensured it would perform its design function as required by licensee procedure NPG-SPP-06.2, Preventive Maintenance.

Description: The 0-FCV-067-0048 valve is a quarter-turn butterfly valve with a motor operator that allowed remote operation of the valve. The valve is the Residual Heat Removal Service Water/Emergency Equipment Cooling Water (RHRSW/EECW) cross-tie valve installed in the discharge piping between the D1 RHRSW pump and the D3 EECW pump. The valve is required to be closed to maintain a boundary between these two interfacing systems. As allowed by system operating instructions, when the valve is open, it aligns the D1 RHRSW pump to the South header of the EECW system. The valve was installed during plant construction to support commercial operation of Unit 1 in August 1974. The licensee determined that the valve had not been replaced since its original installation. While trouble shooting pump performance issues, it was determined that the valve disc was made of cast iron and was subjected to the conditions of the RHRSW system which circulated water from the Tennessee River.

An inspection performed while troubleshooting RHRSW pump performance issues on January 10, 2013, revealed the 0-FCV-067-0048 valve disc separated from the valve stem and the pieces of the valve disc were found in the pipe downstream at the inlet of the 'D' EECW strainer. According to the licensee's root cause report associated with PER 671314, the direct cause of the valve failure was the cumulative effects of its age and the pressure transients in the system. The licensee relied on the requirements of the inservice testing (IST) program for preventive maintenance on this valve. The IST program required remote position indication of the motor operated valve based on the licensee's classification of the valve as a passive valve. Licensee procedure NPG-SPP-06.2, Preventive Maintenance paragraph 3.2.1.B, required preventive maintenance programs to be structured to maintain components in a manner that permits them to perform their design functions. According to BFN-50-7067, General Design Criteria Document for EECW, a design function of the 0-FCV-067-0048 valve is to isolate the EECW system from interfacing systems when necessary so that the EECW system may perform its required nuclear safety function. The inspectors concluded the lack of adequate preventive maintenance prevented identification of valve degradation and the valve was left in service until it catastrophically failed and could not perform a design function of system isolation. This valve failure resulted in unavailability of the D3 EECW pump and required the D1 RHRSW pump be aligned to supply the South EECW header. Additionally, with this valve failed and unable to perform its isolation function credited in the safe shutdown analysis, part of the required EECW system flow would be diverted through the RHRSW heat exchanger during certain fire events.

Analysis: The licensee's failure to establish an adequate preventive maintenance program to maintain the 0-FCV-067-0048 valve in a manner that ensured it would perform its design function as required by licensee procedure NPG-SPP-06.2, Preventive Maintenance, was a performance deficiency. This finding was determined to be more than minor because it was associated with the Protection Against External Events (fires) attribute of the Mitigating Systems cornerstone objective and adversely affected the cornerstone objective to ensure availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the 0-FCV-067-0048 valve failed and could not perform its isolation function credited in the safe shutdown analysis. The inspectors evaluated the significance of the finding using Inspection Manual Chapter (IMC) 0609 Appendix F, "Fire Protection Significance Determination Process" and assigned the finding a Moderate Degradation rating. Since a multiple fire area assessment effort for this type of finding was beyond the intended scope of the fire protection SDP Phase 2 analysis, the finding was forwarded to the senior reactor analysts for review. Because the finding could not be screened as Green, nor its safety significance determined prior to issuing the inspection report, it is being characterized as “To Be Determined (TBD)” pending a significance determination. Since the failed valve was replaced on January 16, 2013, per design change notice (DCN) T40472 and can perform its design function, the finding does not represent an immediate safety concern.

The cause of this finding was directly related to the cross-cutting aspect of Appropriately Coordinating Work Activities in the Work Control component of the Human Performance area, because maintenance activities for 0-FCV-067-0048 were more reactive than preventive. [H.3(b)].

Enforcement: 10 CFR 50, Appendix B, Criterion V, Instruction, Procedures and Drawings, required, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Procedure NPG-SPP-06.2, Preventive Maintenance, was a procedure that prescribed activities affecting quality related to the implementation of the preventive maintenance program. NPG-SPP-06.2 paragraph 3.2.1.B required the preventive maintenance program be structured to maintain components in a manner that permits them to perform their design functions. Contrary to the above, since the original installation of 0-FCV-067-0048, the licensee failed to implement an adequate preventive maintenance program as prescribed by NPG-SPP-06.2, Preventive Maintenance. Specifically, the licensee failed to ensure the preventive maintenance program was structured to maintain 0-FCV-067-0048 in a manner that permitted it to perform its design function in accordance paragraph 3.2.1.B of NPG-SPP-06.2, Preventive Maintenance. This issue was entered in the licensee’s corrective action program as problem event report (PER) 671314. The failed valve was replaced on January 16, 2013, with a new valve with a stainless steel disk. Further corrective actions were planned to develop adequate preventive maintenance activities for this valve. Pending determination of the finding’s final safety significance, this finding is identified as an apparent violation: AV 05000259, 260, 296/2013002-001, Failure to Implement Preventive Maintenance Program.

1R18 Plant Modifications

.1 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the Design Change Notice (DCN) and completed work package (WO 114303359) for T-DCN T40472A/PIC 70944, Replace Valve 0-FCV-067-0048, (RHRSW / EECW crosstie) which included 0-MVOP-67-48 and 0-MTR-67-48. The inspectors reviewed licensee procedures NPG-SPP-09.3, Plant Modifications and Engineering Change Control, and NPG-SPP-06.9.3, Post-Modification Testing, and observed part of the licensee’s activities to implement this design change made while the unit was online. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify that the modifications had not affected system operability or availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents. This activity constitutes one Permanent Plant Modification sample.

b. Findings

No findings were identified.

.2 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification of the release path made to address flooding of the Radiological Waste Building. Inspectors verified regulatory requirements were met, along with procedures such as NPG-SPP-9.3, Plant Modifications and Engineering Change Control; NPG-SPP-9.5, Temporary Alterations; and NPG-SPP-6.9.3, Post-Modification Testing. The inspectors also reviewed the associated 10 CFR 50.59 screening and evaluation and compared each against the UFSAR and TS to verify that the modification did not affect operability or availability of the affected system. Furthermore, the inspectors walked down the modification to ensure that it was installed in accordance with the modification documents and reviewed post-installation and removal testing to verify that the actual impact on permanent systems was adequately verified by the tests. This activity constitutes one Temporary Plant Modification sample.

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors witnessed and reviewed the four post maintenance tests (PMT) listed below to verify that procedures and test activities confirmed SSC operability and functional capability following the described maintenance. The inspectors reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed 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 licensee procedural requirements. Furthermore, the inspectors verified that problems associated with PMTs were identified and entered into the CAP. This activity constituted four Post Maintenance Test inspection samples.

- Post maintenance test of RHRSW Pump D1 crosstie to EECW valve and actuator (0-FCV-067-0048) per T-DCN 40472A/PIC 70944, WO 114303359
- Post maintenance test of Unit 3 Core Spray Loop II Room Cooler fan bearing and discharge and suction valve breakers, WOs 113881247, 113977958, 111615754, 113977981, 113978108, and 113978090
- Post maintenance test of 3B EDG following generator fan bearing replacement WO 112808386
- Post maintenance test of Unit 1 Source Range Monitor 'C' due to replacement of Preamp, WO 114497817

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 2 Scheduled Refueling Outage (U2R17)

a. Inspection Scope

From March 14, 2013, through March 31, 2013, the inspectors examined the initial Unit 2 critical outage activities to verify that they were conducted in accordance with TS, applicable plant procedures, and the licensee's outage risk assessment and management plans. The inspectors also monitored critical plant parameters, and observed operator control of plant conditions during Cold Shutdown (Mode 4) and Refueling (Mode 5). Some of the significant outage activities specifically reviewed and/or witnessed by the inspectors were as follows:

Outage Risk Assessment

Prior to the U2R17 refueling outage that began on March 14, 2013, the inspectors attended outage risk assessment team meetings and reviewed the Outage Risk Assessment Report to verify that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing an outage plan that assured defense-in-depth of safety functions were maintained. The inspectors also reviewed the daily U2R17 Refueling Outage Reports, including the Outage Risk Assessment Management (ORAM) Safety Function Status, and regularly attended the twice per day outage status meetings. These reviews were compared to the requirements in licensee procedure NPG-SPP-07.2, Outage Management, and Technical Specifications. These reviews were also done to verify that for identified high risk significant conditions, due to equipment availability and/or system configurations, contingency measures were identified and incorporated into the overall outage and contingency response plan. Furthermore, the inspectors frequently discussed risk conditions and designated protected equipment with Operations and outage management personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process

The inspectors witnessed the shutdown and cooldown of Unit 2 in accordance with licensee procedures OPDP-1, Conduct of Operations; 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; and 2-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 2-OI-74, Residual Heat Removal System (RHR); 2-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Alternate Decay Heat Removal System Failures; and conducted a main control room panel and in-plant walkdowns of system and components to verify correct system alignment. During planned evolutions that resulted in increased outage risk conditions for shutdown cooling, inspectors verified that the plant conditions and systems identified in the risk mitigation strategy were available. 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 Alternate Decay Heat Removal (ADHR) system.

Critical Outage Activities

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

- Verified Reactor Coolant System (RCS) inventory controls, specifically the makeup during operations with the potential to drain the reactor vessel
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) and TS compliance during the various shutdown modes of operation, and mode transitions
- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations and overall integrity
- Examined foreign material exclusion controls particularly in proximity to and around the reactor cavity, equipment pit, and spent fuel pool
- 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 2-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, the inspectors witnessed fuel handling operations during the two Unit 2 reactor core fuel shuffles performed in accordance with TS and applicable operating procedures, such as 0-GOI-100-3A, Refueling Operations (In Vessel), 0-GOI-100-3B, Operations in the Spent Fuel Pool, and 0-GOI-100-3C, Fuel Movement Operations During Refueling.

Corrective Action Program

The inspectors reviewed PERs generated during U2R17 and attended management review committee (MRC) meetings to verify that initiation thresholds, priorities, mode holds, operability concerns and significance levels were adequately addressed. Resolution and implementation of corrective actions of several PERs were also reviewed for completeness.

b. Findings

No findings were identified.

.2 Unit 3 Forced Midcycle Outage Due To Condenser Circulating Water Leakage

a. Inspection Scope

On February 11, 2013, Unit 3 entered an unplanned forced shutdown following a planned downpower and manual scram. The shutdown was initiated to enact repairs on the Condenser Circulating Water system which was suspected of being a contributor to excessive water in-leakage into the turbine building and subsequent cause of flooding issues with the radwaste system. Operators commenced a restart of Unit 3 on February 20, 2013, which was observed by the Resident Inspectors. The unit entered Mode 2 at 0925 hours followed by Mode 1 at 2231 hours on February 20, 2013 and subsequently synchronized to the grid at 0520 hours on February 21, 2013.

During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment and outage management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of Hot Shutdown (Mode 3) conditions, and critical plant parameters
- Plant Oversight Review Committee (PORC) event review and restart meeting on February 18, 2013
- Reactor startup and power ascension activities per General Operating Instruction (GOI) 3-GOI-100-1A, Unit Startup
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage and attended Management Review Committee meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings were identified.

.3 Unit 3 Forced Outage Due To Loss of Condenser Vacuum

a. Inspection Scope

On February 25, 2013, at 1313 hours, Unit 3 automatically scrammed from 92 percent RTP due to actuation of the reactor protection system as a result of a turbine trip. The cause of the turbine trip was from a loss of condenser vacuum caused by a failure of a pipe in the reactor feedwater long cycle maintenance line connection to a miscellaneous drain header to the condenser. Main steam isolation valves (MSIVs) were manually closed to isolate the leak and no safety relief valves were automatically cycled during the event. All other systems operated as expected. Resident inspectors observed the operators actions following the scram. After repairs were made to the condenser, operators commenced a restart of Unit 3 and the unit entered Mode 2 on February 28, 2013, at 0054 hours followed by Mode 1 at 1322 hours and subsequently synced to the grid at 1744 hours on February 28, 2013. Resident inspectors observed the unit restart.

During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment, including operator's actions and plant responses following the scram. Inspectors also closely followed the repair plans and extent of condition reviews of the event. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of the unit and critical plant parameters to cold Shutdown (Mode 4) conditions
- PORC event review and restart meeting on February 27, 2013
- Reactor startup and power ascension activities per General Operating Instruction (GOI) 3-GOI-100-1A, Unit Startup
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities
- Mode change risk assessment

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage and attended management meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings were identified.

.4 Unit 1 Forced Outage Due To Lowering Condenser Vacuum

a. Inspection Scope

On March 19, 2013, at 0402 hours, Unit 1 was manually scrammed from 95 percent RTP due to lowering main condenser vacuum. The cause of the vacuum loss was due to a leak on a feedwater heater level control line. Operators worked to troubleshoot the issue, however, condenser vacuum continued to degrade and operators recognized the continued trend to the turbine trip setpoint and manually scrammed the reactor prior to automatic RPS initiation of the scram. Following the scram the condenser vacuum stabilized. Main steam isolation valves remained opened and reactor pressure was controlled via the turbine bypass valves. Resident inspectors observed the operators actions following the scram. All equipment responded as designed and as expected. Following condenser repairs operators commenced a restart of Unit 1 and the unit entered Mode 2 on March 27, 2013, at 1114 hours followed by Mode 1 at 0050 hours on March 28, 2013. Subsequently the unit synchronized to the grid at 0540 hours on March 28, 2013. Resident inspectors observed the unit restart. The unit returned to 100 percent RTP on March 29, 2013, at 1506 hours.

During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment, including operator's actions and plant responses following the scram. Inspectors also closely followed the repair plans and extent of condition reviews of the event. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of the unit and critical plant parameters to cold Shutdown (Mode 4) conditions
- PORC event review and restart meeting on March 22, 2013
- Reactor startup and power ascension activities per General Operating Instruction (GOI) 1-GOI-100-1A, Unit Startup
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 1 forced outage and attended management meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions of, and/or reviewed completed test data for the following 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. This activity constituted seven inspection samples, two in-service tests, four routine tests, and one containment isolation test.

In-Service Tests:
- January 20, 2013, RHRSW Pump D3, IST Group A Quarterly Pump Test
- March 12, 2013, Control Bay Chill Water Pump A Augmented Inservice Test

Routine Surveillance Tests:
- January 10, 2013, 0-SI-4.8.A.1-1(a) Liquid Effluent Batch Release (Other than plant radwaste tanks)
- January 25, 2013, 3-SR-3.5.1.6(CS II), Core Spray Flow Rate Loop II
- March 13, 2012, 1/2/3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration
- March 16, 2013, 0-SR-3.8.1.9(B), B Emergency Diesel Load Acceptance Test

Containment Isolation Valve Tests:
- March 16, 2013, 2-SR-3.6.1.3.10(D), As Found Local Leak Rate Test (LLRT) – Main Steam Line D: Penetration X-7D

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

During the report period, the inspectors observed an Emergency Preparedness (EP) Severe Accident Management Guidelines training drill that contributed to the licensee's Drill/Exercise Performance and Emergency Response Organization performance indicator (PI) measures on February 6, 2013. This drill was intended to identify any licensee weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room, Technical Support Center, and Operations Support Center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure, and licensee conformance with other applicable Emergency Plan Implementing Procedures. The inspectors also attended the post-drill critiques to compare any inspector-observed weaknesses with those identified by the licensee in order to verify whether the licensee was properly identifying EP related issues and entering them into the CAP, as appropriate. This activity constituted one inspection sample.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Reactor Coolant System (RCS) Activity and Leakage

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following PIs, including procedure NPG-SPP-02.2, Performance Indicator Program. The inspectors examined the licensee's PI data for the specific PIs listed below for the first through fourth quarters of 2012. 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 review licensee records to verify the PI data was appropriately captured, calculated correctly, and discrepancies were 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. This activity constituted six inspection samples.

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

b. Findings

No findings 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 and Service Request (SR) reports, and periodically attending Corrective Action Review Board (CARB) and PER Screening Committee (PSC) meetings.

.2 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors reviewed the specific corrective actions associated with PER 317464, Technical Specification 5.5.2 which addressed Primary Coolant Sources Outside Containment.

b. Assessment and Observations

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

The licensee created PER 317464 to document corrective actions taken as a result of a NRC Green violation (NCV 05000259, 260, and 296/2011002-03) in response to failing to create and maintain a program to track primary coolant sources outside containment as required by Technical Specification (TS) 5.5.2. This was significant because the licensee had committed to the requirements of General Design Criteria (GDC)-19 when they were permitted to use the Alternate Source Term as described in 10 CFR 50.67. The NRC approved method to comply with this requirement was included in the Improved Technical Specifications (ITS) which were adopted by Browns Ferry which included the primary coolant leakage tracking source program. The immediate corrective action was to develop leakage tables that described the leaks outside containment that were present, the adjusted leak rate of the component, and the repair plan. The extent of condition included all programs added as a result of adopting the ITS. The final corrective action was to develop and maintain the program as required by TS 5.5.2. This program is codified in Technical Instruction 0-TI-578, Minimizing Primary Coolant Sources Outside Containment. Evidence of the execution of this program was observed based on the results of leakage surveillances, documentation of leaks and their respective repair plans in the primary coolant systems outside containment, and interviews with the program managers. Additionally, the licensee took an active role to make improvements in the program to gain additional leakage margin by clarifying calculations and reducing the leakage acceptance criteria to a lower amount. These changes were documented in PER 700517. This activity constituted one Identification and Resolution of Problems Selected Issue inspection sample.

4OA3 Follow-up of Events

.1 (Closed) Licensee Event Report (LER) 05000259/2012-006-02, High Pressure Coolant Injection System Turbine Failed to Trip Using the Manual Trip Push Button

a. Inspection Scope

The inspectors reviewed revised LER 05000259/2012-006-02 dated December 31, 2012. Inspectors reviewed revised information from PER 539040 related to this event. The revised LER provided additional information including; root cause, contributing factors, analysis of the event, extent of condition and assessment of safety consequences which included past operability and the interrelation of past operability with other system availabilities. The licensee's analysis concluded that redundant systems remained operable to maintain safe shutdown capability during the time period that HPCI would have been unable to perform its safety function.

b. Findings

One finding was previously identified and documented in IR 05000259/2012003; Section 4OA7. No additional findings were identified regarding the revised LER. Previous LER revisions 05000259/2012-006-00, and -01 were closed in IR 05000259/2012004. This LER is considered closed.

.2 (Closed) Licensee Event Report (LER) 05000259/2012-010-00, Primary Containment Isolation Valve Inoperable for Longer than Allowed by the Technical Specifications

a. Inspection Scope

The inspectors reviewed the LER for performance deficiencies and violations of regulatory requirements. The LER was associated with a primary containment isolation valve (PCIV) surveillance test failure of an instrument excess flow check valve (EFCV) (1-ECKV-068-0065B) installed on Unit 1. During troubleshooting it was identified that the check valve had been installed incorrectly in a reverse orientation since October 15, 2006. The check valve was in a sensing line for the reactor recirculation system and is designed to reduce flow in the event of a rupture of the line outside of primary containment. TS 3.6.1.3, Primary Containment Isolation Valves (PCIVs), required that PCIVs be operable while Unit 1 was in Modes 1, 2, and 3. The TS action statement C.1 required that the affected flow path be isolated by use of at least one closed and de-activated automatic valve within 12 hours for excess flow check valves (EFCVs). The valve was replaced and verified to be correctly installed via testing.

During the investigation the licensee determined that there was inadequate procedural guidance for installation of these valves. Unit 1 restart testing identified issues with the valve. Due to a human performance error the test indications that could have identified the issue with the valve installation were incorrectly evaluated as acceptable. An extent of condition review was performed on related EFCVs through past surveillances and it was determined that the incorrect installation and associated human performance analysis error was an isolated event.

The inspectors reviewed PERs, WOs and the root cause report associated with this event and discussed the issue with appropriate members of plant staff. This condition was documented in the licensee's corrective action program as PER 646600. Additional documents reviewed are listed in the Attachment.

b. Findings

The enforcement aspects of this finding are discussed in Section 4OA7. This LER is closed.

.3 (Closed) Licensee Event Report (LER) 050000260/2012-006-00, Automatic Reactor Scram Due to Loss of Power to the Reactor Protection System

a. Inspection Scope

On December 22, 2012, Unit 2 automatically scrammed from approximately 100 percent power when a Unit Supervisor performed an error in implementing Operating Instruction guidance during response to a loss of 2B Reactor Protection System (RPS). The operator incorrectly opened the supply breaker on the in-service 2A RPS bus motor generator set bus which resulted in a reactor scram logic and containment isolation. The inspectors reviewed the applicable LER that was issued on February 20, 2013, and its associated PER 660862, which included the root cause analysis (RCA) and corrective actions. One finding associated with this LER review is discussed below.

b. Findings

This LER is considered closed with one finding identified.

Introduction: A self-revealing AV of Technical Specification 5.4.1 was identified for the licensee's failure to properly implement procedure 2-OI-99, Reactor Protection System. Specifically, during restoration of 2B Reactor Protection System 480 volt power, the RPS motor generator set tie to battery BD 2 Breaker on the 2A RPS bus motor generator set was incorrectly opened.

Description: On December 22, 2012, during a test of the capability to parallel the 3D and D EDGs, a malfunction caused the two diesels' load sharing function to not work properly and at 1134 hours, the 3D EDG attempted to reverse power the D EDG. This resulted in loss of the D 4kv shutdown board and the 2B RPS bus. The 2B RPS bus loss also caused actuation of primary containment isolation system (PCIS) groups 2, 3, 6, and 8 and a half-trip condition on PCIS group 1. This resulted in a loss of reactor building ventilation which caused a steady increase in main steam vault temperatures. By design, at 189 degrees in the main steam vault, the main steam isolation valves (MSIVs) isolate and a reactor scram signal is generated. Knowledge of this feature imposed time pressure on the operators to restore the 2B RPS bus. An operator was dispatched to restart the RPS bus motor generator set and re-energize the 2B RPS bus per procedure 2-OI-99, Reactor Protection System, Section 5, STARTUP.

A pre-job brief was not conducted and no operator was sent to perform peer checking of the performance of the task. Additionally, a potential human error trap existed in procedure 2-OI-99, step 5.1 [3], in that, it referenced both A and B RPS motor generator set breakers. The listed breakers were only separated by a parenthesis to designate the B train breakers. At time 1152 hours, eighteen minutes after loss of 2B RPS bus, the operator incorrectly opened the RPS motor generator set tie to battery BD 2 Breaker to the 2A RPS motor generator set which de-energized the 2A RPS bus causing a Unit 2 reactor scram and MSIV closure. This also resulted in a loss of main condenser vacuum and loss of main feedwater. The licensee root cause analysis found the direct cause of the event was that the operator, an SRO qualified Unit Supervisor, did not perform the act of "self-checking" correctly. The licensee root cause was determined to be the use of supervisors performing plant manipulations instead of the Unit Operators or Assistant Unit Operators.

Analysis: The licensee's failure to properly implement procedure 2-OI-99, Reactor Protection System when an operator incorrectly opened the RPS motor generator set tie to battery BD 2 Breaker on the 2A RPS bus motor generator set, was a performance deficiency. This finding was determined to be greater than minor because it was associated with the human performance attribute of the initiating events cornerstone, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. Specifically, opening the RPS motor generator set tie to battery BD 2 Breaker on the 2A RPS bus motor generator set directly resulted in an automatic reactor scram and MSIV closure. The inspectors evaluated the significance of the finding using Inspection Manual Chapter 0609, Appendix A, Phase 1 for At-Power Significance Determination Process and determined that a detailed risk evaluation was required because it contributed to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. Because the finding could not be screened as Green, nor its safety significance determined prior to issuing the inspection report, it is being characterized as "To Be Determined (TBD)," pending a significance determination. The finding does not present an immediate safety concern because the licensee has taken actions to prevent recurrence of the associated human performance error that caused the event.

The cause of this finding was directly related to the cross-cutting aspect of Human Error Prevention in the Work Practices component of the Human Performance area, because the lack of adequate self-check, peer checking, and pre-job briefing resulted in the operator opening the incorrect breaker. [H.4(a)]

Enforcement: Technical Specification 5.4.1.a, Procedures, required that written procedures shall be established, implemented, and maintained covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. The procedure for Startup, Operation, and Shutdown of the Reactor Protection System was listed as a recommended procedure in section 4.y of Regulatory Guide 1.33, Appendix A. Contrary to the above, on December 22, 2012, the licensee did not properly implement a procedure recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Specifically, the licensee failed to properly implement the procedure for Startup, Operation, and Shutdown of the Reactor Protection

System, 2-OI-99, Reactor Protection System, step 5.1[3], when an operator incorrectly opened the RPS motor generator set tie to battery BD 2 Breaker on the A RPS bus motor generator set while attempting to start the B RPS bus motor generator set. The failure to properly implement 2-OI-99 caused a Unit 2 reactor scram and MSIV closure. The licensee took immediate actions to respond to the resultant Unit 2 scram and placed the unit in a shutdown condition. Subsequent corrective actions included operator training and procedure revisions. This issue was entered in the licensee's corrective action program as PER 660862. Pending determination of the finding's final safety significance, this finding is identified as an apparent violation: AV 05000260/2013002-02, Failure to Follow Operating Procedure Guidance Resulted in Unit 2 Reactor Scram.

.4 (Closed) Licensee Event Report (LER); 05000296/2012-004-01, Manual Reactor Scram During Startup Due to Multiple Control Rod Insertion

a. Inspection Scope

The inspectors reviewed the LER for potential performance deficiencies and/or violations of regulatory requirements. The LER was associated with the Unit 3 manual reactor scram that occurred during a reactor startup on May 24, 2012. The inspectors reviewed the root cause report revisions associated with this event, the plant's corrective actions that have occurred or are planned, and discussed the issue with appropriate members of the Browns Ferry Nuclear Plant staff. This condition was documented in the licensee's corrective action program as PER 558437. LER 05000296/2012-004-00 was closed previously in NRC integrated inspection report 05000259, 260, 296/2012005. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified. These LERs are closed.

.5 (Closed) Licensee Event Report (LER) 05000260/2012-004-00, High Pressure Coolant Injection System Rendered Inoperable Due to an Inadvertent Actuation of Primary Containment Isolation System

a. Inspection Scope

On August 17, 2012, Unit 2 High Pressure Coolant Injection (HPCI) system logic isolated a steam line during testing. Operators declared the system inoperable and entered TS actions to verify that the Reactor Core Isolation Cooling (RCIC) system was operable. The HPCI system isolated due to a spurious indication on the steam line space high temperature B channel switch during simultaneous testing on the opposing A channel. Corrective actions were taken to replace all the steam line space high temperature B channel switches and reviews were made of other applicable system switches for extent of condition. After successful testing the HPCI system was returned to service the following day.

The inspectors reviewed PERs, surveillance procedures, WOs and the root cause evaluation report associated with this event. This condition was documented in the licensee's corrective action program as PER 596706. Additional documents reviewed are listed in the Attachment.

b. Findings

No findings were identified. This LER is closed.

.6 (Closed) Licensee Event Report (LER) 296/2011-003-01, Automatic Reactor Scram Due to a Main Turbine Generator Load Reject

a. Inspection Scope

The inspectors reviewed Revision 1 of the LER dated November 26, 2012. This revised LER was submitted to provide the results of the licensee's completed investigation and revised causal analysis. The initial follow-up of this event by the inspectors was documented in Section 4OA3.10 of IR 05000296/2011004. The original LER 296/2011-003-00, dated November 28, 2011, and applicable PER 440539, including cause determination and corrective action plans, were reviewed by the inspectors and documented in Section 4OA3.1 of NRC IR 05000296/2012003 with no findings identified.

On September 28, 2011, Unit 3 automatically scrammed from 100 percent power due to a power to load unbalance (i.e., main generator load reject) automatic trip of the main turbine generator caused by a broken debris screen. The licensee concluded that the direct cause of the Unit 3 turbine trip and scram was the isolated-phase bus C debris screen failure.

b. Findings

No findings were identified. This LER is closed.

.7 (Closed) Licensee Event Report (LER) 50-296/2012-003-01, Browns Ferry Nuclear Plant Unit 3 Automatic Reactor Scram due to De-Energization of Reactor Protection System from Actuation of 3A Unit Station Service Transformer Differential Relay

a. Inspection Scope

The inspectors reviewed Revision 1 of the LER dated November 26, 2012. This revised LER was submitted to provide the results of the licensee's completed investigation and revised causal analysis. The initial follow-up of this event by inspectors was documented in Section 4OA3.1 of NRC IR 05000296/2012003. The original LER 50 296/2012-003-00, dated July 23, 2012, and applicable PER 555573, including cause determination and corrective action plans, were reviewed by the inspectors and documented in Section 4OA3.2 of NRC IR 05000296/2012004, where a self-revealing finding (FIN) was identified for the licensee's failure to provide an adequate design review of vendor calculations, FIN 05000296/2012004-02, Automatic Reactor Scram Due to Inadequate Design Review of Relay Setting. On May 22, 2012, Unit 3 automatically scrammed from approximately 19 percent power when the premature actuation of the 3A Unit Station Service Transformer (USST) differential current protection relay caused a loss of the 500KV offsite power system.

The inspectors verified that the supplemental information provided in the revised LER was complete and accurate and that the additional information was not of a significant nature to warrant a change to the original LER disposition. No additional licensee performance deficiency was identified by the inspectors.

b. Findings

No additional findings were identified. This LER is closed.

.8 (Closed) Licensee Event Report (LER) 50-296/2012-005-01, Automatic Reactor Scram Due to an Actuation of a Main Transformer Differential Relay

a. Inspection Scope

The inspectors reviewed Revision 1 of the LER dated November 29, 2012. This revised LER was submitted to provide the results of the licensee's completed investigation and revised causal analysis. The initial follow-up of this event by inspectors was documented in Section 4OA3.3 of NRC IR 05000296/2012003. The original LER 50 296/2012-005-00, dated July 30, 2012, and applicable PER 558183, including cause determination and corrective action plans, were reviewed by the inspectors and documented in Section 4OA3.4 of NRC IR 05000296/2012004, where a self-revealing finding (FIN) was identified for the licensee's failure to adequately test a Unit 3 main turbine generator current transformer, FIN 05000296/2012004-03, Automatic Reactor Scram Due to Inadequate Testing of Current Transformer. On May 29, 2012, Unit 3 automatically scrammed from approximately 75 percent power when the premature actuation of the main transformer differential over-current protection relay caused a main generator load rejection signal.

The inspectors verified that the supplemental information provided in the revised LER was complete and accurate and that the additional information was not of a significant nature to warrant a change to the original LER disposition. No additional licensee performance deficiency was identified by the inspectors.

b. Findings

No additional findings were identified. This LER is closed.

.9 (Closed) Licensee Event Report (LER) 296/2013-001-00, Inoperable Emergency Diesel Generators due to Failed or Degraded Electric Generator Casing Fan Bearings

a. Inspection Scope

The inspectors reviewed the LER, dated March 11, 2013, and the associated PER 665217, including the root cause analysis, operability determinations, and corrective action plans. On January 9, 2013, while performing operator rounds near the Unit 3, 3D EDG, the licensee discovered metal residue and grease around the generator blower shaft. The licensee determined the generator blower inboard bearing (coupling side) had failed during a previous post maintenance test, as verified by licensee vibration data, rendering the 3D EDG inoperable. Following return to service of the 3D EDG and extent-of-condition inspections, the licensee determined that two additional Unit 3 EDGs had blower bearings that were degraded but not failed, and were also determined to be inoperable. The licensee concluded that the direct cause of the 3D EDG bearing failure was the absence of lubrication to the internal parts of the EDG blower bearing due to age related breakdown of the grease. The licensee determined two root causes to be inadequate component level assessment of the blower shielded bearings for failure modes and impacts and ineffective industry vibration monitoring standards. All four Unit 3 EDG generator blower bearings were replaced.

b. Findings

The enforcement aspects of this finding are discussed in Section 4OA7. This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Temporary Instruction 2515/187 – Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors verified that licensee’s walkdown packages (CTP-FWD-100, Appendix B Walkdown Record Forms for Work Order 113618794 and Flood Protection Feature IDs 0-DOOR-260-272, 0-DOOR-260-800, 0-DOOR-260-A-RHRSW, 0-DOOR-260-B-RHRSW, 0-DOOR-260-C-RHRSW, and 0-DOOR-260-D-RHRSW) contained the elements as specified in NEI 12-07, Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features.

The inspectors accompanied the licensee on their walkdown of the Common Unit Reactor Building Flood Gate on August 7, 2012, and verified that the licensee confirmed the following flood protection features:

- Manual actions to ensure the Reactor Building Flood Gate could be operated within the required time considering the design basis flood
- Adequate consumables to support the credited Reactor Building Flood Gate
- Procedures to operate the credited Reactor Building Flood Gate

The inspectors independently performed their walkdown and verified that hatches and manhole covers in the Residual Heat Removal Service Water (RHRSW) Pump Rooms were in place and confirmed the following flood protection features:

- Visual inspection for indications of degradation that would prevent credited functions from being performed
- Critical dimensions
- Available physical margin
- Flood protection functionality

The inspectors verified that noncompliance's with current licensing requirements, and issues identified in accordance with the 10 CFR 50.54(f) letter, Item 2.g of Enclosure 4, were entered into the licensee's corrective action program. In addition, issues identified in response to Item 2.g that could challenge risk significant equipment and the licensee's ability to mitigate the consequences will be subject to additional NRC evaluation.

b. Findings

No NRC-identified or self-revealing findings were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

Exit meetings covering the first quarter integrated resident report were conducted on April 5, 2013, and April 26, 2013, with members of the licensee management staff. All proprietary information that was provided to the inspectors was returned to the licensee.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and area violations of NRC requirements which meet the criteria of the NRC Enforcement Policy, for being dispositioned as a Non-Cited Violation.

- Technical Specifications 3.6.1.3, Primary Containment Isolation Valves (PCIVs), required that PCIV's be operable while Unit 1 was in Modes 1, 2, and 3. The Technical Specification (TS) action statement C.1 required that the affected flow path be isolated by use of at least one closed and de-activated automatic valve within 12 hours for excess flow check valves (EFCVs). Contrary to the above, between May 21, 2007, and November 27, 2012, excess flow check valve 1-ECKV-068-0065B was installed in a reverse orientation, preventing the valve from performing its function to reduce flow downstream in the event of a line rupture outside of primary containment, and the affected flow path had not been isolated.

  The finding was screened in accordance with IMC 0609 Appendix H, Containment Integrity SDP and was characterized to be of very low safety significance (Green) because the valve was a one-inch valve and would not generally contribute to Large Early Release Frequency (LERF) as discussed in IMC 0609, Appendix H.

- Unit 3 Technical Specification 3.3.8.1, AC Sources - Operating, required Emergency Diesel Generators (EDGs) to be operable in Modes 1, 2, and 3, and with multiple EDGs inoperable, required all but one EDG be returned to service in 2 hours or be in Mode 3 within 12 hours and in Mode 4 within 36 hours. Contrary to this, between December 22, 2012, and January 9, 2013, the licensee determined that multiple EDGs were inoperable as a result of failed 3D EDG and degraded 3A and 3B EDG generator blower bearings. This TS violation was entered into the licensee's CAP as PERs 665217, 675339, and 675952. This finding represented an actual loss of function of the 3D EDG for greater than the Technical Specification allowed outage time, and therefore, required a detailed risk evaluation. The regional Senior Reactor Analyst performed an analysis of the finding. The dominant results were loss of offsite power sequences with common cause EDG failure combinations that result in injection and suppression pool cooling loss of function. Tornado and Seismic were not major contributors. The risk impact was determined to be very low safety significance (Green) for all three units.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
C. Bailey, ICS System Engineer
J. Blenkinsopp, Refuel Floor Supervisor
J. Colvin, Engineering Manager
J. Douglass, Mechanical Analysis
M. Ellet, BFN Maintenance Rule Engineer
J. Emens, Nuclear Site Licensing Manager
F. Forscello, ISI/ISO
J. Guthrie, Diesel Generator System Engineer
E. Johnson, Diesel Generator System Engineer
F. Nilsen, Site Engineer ISI/NDE
K. Polson, Site Vice President
C. Reischman, Diesel Generator Battery Engineer
R. Smith, System Engineer
S. Wentzel, System Engineer

NRC Personnel

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000259, 260, 296/2013002-01	AV	Failure to Implement Preventive Maintenance Program, (Section 1R15)
05000260/2013002-02	AV	Failure to Follow Operating Procedure Guidance Resulted in Unit 2 Reactor Scram (Section 4OA3.3)

Closed

05000259/2012-006-02	LER	High Pressure Coolant Injection System Turbine Failed to Trip Using the Manual Trip Push Button (Section 4OA3.1)
05000259/2012-010-00	LER	Primary Containment Isolation Valve Inoperable for Longer than Allowed by the Technical Specifications (Section 4OA3.2)

05000260/2012-004-00	LER	High Pressure Coolant Injection System Rendered Inoperable Due to an Inadvertent Actuation of Primary Containment Isolation System (Section 4OA3.5)
050000260/2012-006-00	LER	Automatic Reactor Scram Due to Loss of Reactor Protection System (Section 4OA3.3)
05000296/2012-004-01	LER	Manual Reactor Scram During Startup Due to Multiple Control Rod Insertion (Section 4OA3.4)
05000296/2011-003-01	LER	Automatic Reactor Scram Due to a Main Turbine Generator Load Reject (Section 4OA3.6)
05000296/2012-003-01	LER	Browns Ferry Nuclear Plant Unit 3 Automatic Reactor Scram Due To De-Energization of Reactor Protection System From Actuation of 3A Unit Station Service Transformer Differential Relay (Section 4OA3.7)
05000296/2012-005-01	LER	Automatic Reactor Scram Due to an Actuation of a Main Transformer Differential Relay (Section 4OA3.8)
05000296/2013-001-00	LER	Inoperable Emergency Diesel Generators due to Failed or Degraded Electric Generator Casing Fan Bearings (Section 4OA3.9)

<u>Discussed</u>

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection – Severe Weather Readiness and External Flooding

Dow Corning Reports dated 10/1/1984 and 2/18/1985
DWG 0-47W391-9, Fire Protection – 10CFR50, App. R Penetration, Internal Conduit Fire Seals, Rev. 3
DWG 2-45B891-3, Conduit & Grounding Cable – Conduit Seal Misc Matl Method, Rev. 1
PER 623106, Potential Non-conservative Assumptions in Calculations for Leakage in RHRSW Pump Room Sump
PER 666222, Issue Associated with RHRSW and Flooding
PER 669508, Temporary Alteration Identified in A RHRSW Pump Room
PER 671475, Seals for Electrical Penetrations
PER 671475, Seals for Electrical Penetrations
Prompt Determination of Operability for PER 666222

Section 1R04: Equipment Alignment

0-SI -4.8.A.1-1, Liquid Effluent Permit
0-SI-4.5.C.1(1), RHRSW and EECW System Valve Operability Test Data from:
0-SI-4.8.A.1-1(a), Liquid Effluent Batch Release (Other than plant radwaste tanks)
0-SSI-001, Safe Shutdown Instructions, Rev. 0014
0-SSI-10 U-2, 480V Shutdown Board 2A Room
0-SSI-1-1 U-1, RX BLDG FIRE EL 519 through 565 West of Column Line R4
0-SSI-11 U-2, 480V Shutdown Board 2B Room
0-SSI-1-2 U-1, RX BLDG FIRE EL 519 through 565 East of Column Line R4
0-SSI-12 U-3, 480V RMOV Board Room 3B
0-SSI-1-3 U-1, RX Bldg Fire EL 593 North of Column Line R
0-SSI-13 U-3, 480V RMOV Board Room 3A
0-SSI-1-4 U-1, Rx Bldg Fire El 593 South of Column Line R and RHR HX Rooms From EL 565 through 593
0-SSI-14 U-3, 480V Shutdown Board 3A Room
0-SSI-1-5 U-1, Rx Bldg Fire El 621 and 639 North of Column Line R
0-SSI-15 U-3, 480V Shutdown Board 3B Room
0-SSI-16, Control Bldg FIRE EL 593 through EL 617
0-SSI-1-6 U-1, Rx Bldg Fire El 639 South of Column Line R
0-SSI-17 U-1 Battery and Battery Board Room
0-SSI-18 U-2 Battery and Battery Board Room
0-SSI-19 U-3 Battery and Battery Board Room
0-SSI-20 U-1 & 2 Diesel Generator Bldg
0-SSI-2-1 U-2 Rx Bldg Fire El 519 through 565 West of Column Line R11
0-SSI-21 U-3 Diesel Generator Bldg
0-SSI-2-2 U-2 RX Bldg Fire El 519 through 565 East of Column Line R11
0-SSI-22 U-3, 4KV Shutdown Board Rooms 3EA & 3EB
0-SSI-2-3 U-2 Rx Bldg Fire North of Column Line R El 593
0-SSI-23 U-3, 4KV Shutdown Board Rooms 3EC & 3ED
0-SSI-2-4 U-2 Rx Bldg Fire El 593 South of Column Line R and RHR HX Rooms from El 565 through 593
0-SSI-24 U-3, 4KV Bus Tie Board Room
0-SSI-2-5 U-2 Rx Bldg Fire El 621 & 639 North of Column Line R

0-SSI-25-1 Intake Pumping Station El 550’, CCW pump Deck El 565’, Cable Tunnel to Fire Door 440, RHRSW Pump Room B, RHRSW Pump Room D
0-SSI-25-2 RHRSW Pump Room A
0-SSI-25-3 RHRSW Pump Room C
0-SSI-26 Turbine Bldg, Turbine Bldg Side of Cable Tunnel to Fire Door 440, Radwaste Bldg
0-SSI-2-6 U-2 Rx Bldg Fire El 639 South of Column Line R
0-SSI-3-1 U-3 RX BLDG Fire EL 519 thru 565 West of R18, Equipment Hatch between Col R15 and R17, T & U line at EL 593 & 621, EL 639 South of R line
0-SSI-3-2 U-3 Rx Bldg Fire EL 519 thru 565 East of R18
0-SSI-3-3 U-3 RX BLDG Fire EL 593 and RHR HX Rooms
0-SSI-3-4 U-3 RX BLDG Fire EL 621 & EL 639 North of R LINE
0-SSI-4 U-1, 4KV ELECTRIC Board Room 1B
0-SSI-5 U-1, 4KV ELECTRIC Board Room 1A
0-SSI-6 U-1, 480V Shutdown Board Room 1A Room
0-SSI-7 U-1, 480V Shutdown Board 1B Room
0-SSI-8 U-2, 4KV ELECTRIC Board Room 2B
0-SSI-9 U-2, RX BLDG FIRE 4KV ELECTRIC Board Room 2A
0-TI-362(BASES) Inservice Testing (IST) Program Bases Document changes 2008 to 8/21/2010, Rev. 0000
1-47E812-1 Flow Diagram High Pressure Coolant Injection System (Unit 3)
1-47E812-2 Flow Diagram HPCI Oil System (Unit 3)
2-SI-4.5.C.1(3), RHRSW 2012 pump and header operability and flow tests
3-OI-73 High Pressure Coolant Injection (HPCI) System
3-OI-73/ATT-1 High Pressure Coolant Injection System Attachment 1 Valve Lineup Checklist
3-OI-73/ATT-2 High Pressure Coolant Injection System Attachment 2 Panel Lineup Checklist
3-OI-73/ATT-3 High Pressure Coolant Injection System Attachment 3 Electrical Lineup Checklist
3-OI-73/ATT-4 High Pressure Coolant Injection System Attachment 4 Instrument Inspection Checklist
Apparent Cause Evaluation for PER 635729
DCN 40472 disc material change request form for 0-FCV-67-48
DWG 1-47E858-1 Unit 1 Flow Diagram RHRSW System
DWG 1-47E858-1-ISI ASME Section XI RHR Service Water System Code Class Boundaries
DWG 1-47E859-1 Unit 0 and 1 Flow Diagram EECW System
DWG 1-47E859-1-I Unit 0 and 1 Flow Diagram EECW System
DWG 2-47E858-1 Unit 2 Flow Diagram RHRSW System
DWG 2-47E859-1 Unit 2 Flow Diagram EECW System
DWG 3-47E858-1 Unit 3 Flow Diagram RHRSW System
DWG 3-47E859-1 Unit 3 Flow Diagram EECW System
Final Safety Analysis Report 6.4 High Pressure Coolant Injection System
Functionality Evaluation for PER 635729 RHRSW Header low pressure alarm received during 2-SI-4.5.C.1(2) EECW Pump Operation
Maintenance Strategy for BFN-0-FCV-067-0048 dated 2/20/2013
NPG-SPP-09.18.2 Equipment Reliability Classification, Rev. 0001
PER 635729
PER 659017
PER 666247
PER 940127
PER 957689

Technical Specification (TS) 3.5 Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System
Temporary System Drawing – Frac tank release path
Verification and Validation Appendix R Manual Actions for 0-SSI-2-4
WO-94-020448-000
WO-97-011400-000
WO-97-011425-000
0-OI-18 Fuel Oil System, Rev 53
0-OI-82 Standby Diesel Generator Operations
0-OI-82/ATT-1B Standby Diesel Generator B Valve Lineup Checklist, Rev 0101
0-OI-82/ATT-2B Standby Diesel Generator B Panel Lineup Checklist, Rev 0101
0-SR-3.8.1.1 Diesel Generator Fuel Oil Quantity, Rev 0005
DWG 0-47E861-2A Flow Diagram EDG B Starting Air
DWG 0-47E861-5 Flow Diagram EDG B Lube Oil System
DWG 0-47E840-3 Flow Diagram Fuel Oil System
Calculation MD-Q0018-8701641 Diesel Fuel Oil Consumption and 7-day tank sizing, Rev 5
Calculation MD-Q0000-1820080001 Ultra Low Sulfur Diesel Fuel (ULSD) Evaluation, Rev 0
Calculation R14 970521 101 EDG 7 day tank level instrument calibration
NRC Information Notice (IN) 2006-22 New Ultra Low Sulfur Diesel Fuel Could Adversely Impact Diesel Performance
System Health Report, Emergency Diesel Generators
Problem Evaluation Report (PER) 206312 Oil Sample not taken within 1 hour after EDG Shutdown
PER 215296 EDG B Left Bank Air Compressor out of service
PER 245804 EDG B Low oil level during 0-SR-3.8.1.1(B)
PER 605565 EDG B wrong corrosion control chemical used in jacket water system
PER 691830 EDG 125 VDC Battery Chargers need to be replaced
PER 702210 EDG 7 Day Fuel Tank Level calculation inconsistencies
PER 671486 EDG Fuel injectors failed pop test
Event Number 48844 Part 21 Notification of EDG Fuel injectors failing pop test

<u>Section 1R05: Fire Protection</u>
Fire Protection Report, Vol. 1, Fire Hazards Analysis, Rev. 14
Fire Protection Report, Vol. 2, Appendix Q, Pre-Fire Plans, Rev. 49
Fire Protection Report Volume 1, Fire Hazards Analysis for Fire Area 20, Rev 14
Fire Protection Report Volume 2, Appendix U, Pre-Fire Plans for Diesel Generator Building Unit 1 and 2, Rev 0049
Fire Protection Report Volume 1, Fire Hazards Analysis, Section 2, Fire Area 22, Rev. 14
Fire Protection Report Volume 2, Appendix V, Section IV, Pre-Fire Plans for Browns Ferry Nuclear Plant – Diesel Generator Building, Unit 3, Rev. 50
Fire Drill Evaluation Report, Structural, Drill # 00076630, dated 3/13/2013

<u>Section 1R06: Flood Protection Measures</u>
BFN-50-C-7101 Protection from Wind, Tornado Wind, Tornado Depressurization, Tornado Generated Missiles, and External Flooding
Final Safety Analysis Report (FSAR) Appendix 2.4A
Individual Plant Examination For External Events (IPEEE) Internal Fires, High Winds, Floods, Transportation and Nearby Accidents

Section 1R08: Inservice Inspection Activities (71111.08G)

PER 356250, Incorporation of ASME Code Cases into ISI Program needs improvement, 04/18/2011
PER 370184, Through Wall Leak on Emergency Equipment Cooling Water (EECW), 05/18/2011
PER 443133, Inadequate NDE, 10/6/2011
SR 702176, Ultrasonic Data Sheet Discrepancies, 03/26/2013
SR 702178, UT Couplant Bottles without labels, 03/26/2013
2-SI-4.6.G, Browns Ferry Nuclear Plant Unit 2 Inservice Inspection and Risk-Informed Inspection Program Procedure, Rev 50
NPG-SPP-02.3, Revision 5, Operating Experience Program Procedure
N-UT-64, Rev 12, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds
IEP-200, Rev 12, Qualification and Certification for TVA Inspection Services Organization (ISO) Nondestructive Examination (NDE) Personnel
IEP-300, Rev 4, Qualification and Certification of Ultrasonic Inspection Services Organization (ISO) Personnel for Preservice and Inservice ASME Section XI Examinations
MMDP-9, Qualification and Certification of Personnel Performing Welding Processes
TVA Welder/Welding Operator Performance Qualification Records
ASME/ANSI Detail Welding Procedure Specification (DWPS), August 30, 2004, Manual Gas Tungsten Arc Welding (GT)
Work Order 2-SI-4.6.G, Ultrasonic Examination of SLCS N-10-1, Standby Liquid Control System
Work Order 113122325, Ultrasonic Examination of CSS DCS-2-13A, Core Spray System
Work Order 112264638, North Header Supply to U2 RBCCW Drain Valve
Apparent Cause Evaluation Report (Lower Tier), Incomplete UT Examinations, PER 443133, 10/21/2011
QA Record L29871029802, Browns Ferry Nuclear Plant Weld Rod Certification Approval for ER-410
IHI Southwest Technologies, INC Certificates of Qualification
TVA, Record of Liquid Penetrant Exam for Component RHR-2-037-033 COR0, 03/27/13
TVA, Visual Examination (VT-3) of 2-47B455H0066, High Pressure Coolant Injection System Variable Support
CRP-ENG-F-10-010, Browns Ferry Nuclear (BFN) Units 1, 2, and 3, ISI Programs and Watts Bar Nuclear Unit 1 ISI Program, Focused/Snapshot Self-Assessment Report, July 6 through August 13, 2010
Certificate of Compliance, Serial No: E0342, Code Edition ASME Section III, 1995 Edition, 1996 Addenda, Class 1

Section 1R11: Licensed Operator Requalification

NPG-SPP-17.8.1, Licensed Operator Requalification Examination Development and Implementation, Rev. 07
NPG-SPP-17.8.3, Simulator Exercise Guide Development and Revision, Rev. 02
3-SR 3.4.9.1(1)
3-GOI-100-1A Rev 0102
0-TI-464 - U3 RCP 130220-000 Reactivity Control Plan

Section 1R12: Maintenance Effectiveness

0-TI-346 Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting Rev 44
Problem Evaluation Report (PER) 697263 Rod Worth Minimizer (RWM) Failure during Diesel Generator testing
PER 681531 Develop Maintenance rule criteria for RWM

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

BFN Plan of the Day, January 29, 2013
EOOS Software and Operators input for January 29, 2013
BFN Daily Schedule / Work Week 1304, January 28th – February 3, 2013
NPG-SPP-07.3, Work Activity Risk Management Process, Rev. 11
NPG-SPP-09.11.1, Equipment Out of Service (EOOS) Management, Rev. 5
NPG-SPP-09.11.2, Risk Assessment Methods for Technical Specifications, Rev. 0
TS LCOTR 3.0.4 Mode Change Risk Assessment, dated 2/27/2013
2-POI-200.5 Operations with Potential for Draining the Reactor Vessel (OPDRV)
U2R17 Outage Risk Assessment Report Attachment E (Inventory Control during an OPDRV for Control Rod Drive Replacement)

Section 1R15: Operability Evaluations

0-SI-4.5.C.1(1), RHRSW and EECW System Valve Operability Test Data from:1/7/12, 5/26/12, 6/7/12, 8/16/12, 10/11/11, 11/1/12, 11/6/12, 11/22/12, 11/23/11, 12/17/12, 12/18/12, 12/19/12
0-SSI-001 Safe Shutdown Instructions, Rev. 0014
0-SSI-10 U-2, 480V Shutdown Board 2A Room
0-SSI-1-1 U-1 RX BLDG FIRE EL 519 through 565 West of Column Line R4
0-SSI-11 U-2, 480V Shutdown Board 2B Room
0-SSI-1-2 U-1 RX BLDG FIRE EL 519 through 565 East of Column Line R4
0-SSI-12 U-3, 480V RMOV Board Room 3B
0-SSI-1-3 U-1 RX Bldg Fire EL 593 North of Column Line R
0-SSI-13 U-3, 480V RMOV Board Room 3A
0-SSI-1-4 U-1 Rx Bldg Fire El 593 South of Column Line R and RHR HX Rooms From EL 565 through 593
0-SSI-14 U-3, 480V Shutdown Board 3A Room
0-SSI-1-5 U-1 Rx Bldg Fire El 621 and 639 North of Column Line R
0-SSI-15 U-3, 480V Shutdown Board 3B Room
0-SSI-16 Control Bldg FIRE EL 593 through EL 617
0-SSI-1-6 U-1 Rx Bldg Fire El 639 South of Column Line R
0-SSI-17 U-1 Battery and Battery Board Room
0-SSI-18 U-2 Battery and Battery Board Room
0-SSI-19 U-3 Battery and Battery Board Room
0-SSI-20 U-1 & 2 Diesel Generator Bldg
0-SSI-2-1 U-2 Rx Bldg Fire El 519 through 565 West of Column Line R11
0-SSI-21 U-3 Diesel Generator Bldg
0-SSI-2-2 U-2 RX Bldg Fire El 519 through 565 East of Column Line R11
0-SSI-22 U-3, 4KV Shutdown Board Rooms 3EA & 3EB
0-SSI-2-3 U-2 Rx Bldg Fire North of Column Line R El 593
0-SSI-23 U-3, 4KV Shutdown Board Rooms 3EC & 3ED
0-SSI-2-4 U-2 Rx Bldg Fire El 593 South of Column Line R and RHR HX Rooms from El 565 through 593

0-SSI-24 U-3, 4KV Bus Tie Board Room
0-SSI-2-5 U-2 Rx Bldg Fire El 621 & 639 North of Column Line R
0-SSI-25-1 Intake Pumping Station El 550', CCW pump Deck El 565', Cable Tunnel to Fire Door 440, RHRSW Pump Room B, RHRSW Pump Room D
0-SSI-25-2 RHRSW Pump Room A
0-SSI-25-3 RHRSW Pump Room C
0-SSI-26 Turbine Bldg, Turbine Bldg Side of Cable Tunnel to Fire Door 440, Radwaste Bldg
0-SSI-2-6 U-2 Rx Bldg Fire El 639 South of Column Line R
0-SSI-3-1 U-3 RX BLDG Fire EL 519 thru 565 West of R18, Equipment Hatch between Col R15 and R17, T & U line at EL 593 & 621, EL 639 South of R line
0-SSI-3-2 U-3 Rx Bldg Fire EL 519 thru 565 East of R18
0-SSI-3-3 U-3 RX BLDG Fire EL 593 and RHR HX Rooms
0-SSI-3-4 U-3 RX BLDG Fire EL 621 & EL 639 North of R LINE
0-SSI-4 U-1, 4KV ELECTRIC Board Room 1B
0-SSI-5 U-1, 4KV ELECTRIC Board Room 1A
0-SSI-6 U-1, 480V Shutdown Board Room 1A Room
0-SSI-7 U-1, 480V Shutdown Board 1B Room
0-SSI-8 U-2, 4KV ELECTRIC Board Room 2B
0-SSI-9 U-2, RX BLDG FIRE 4KV ELECTRIC Board Room 2A
0-TI-362(BASES) Inservice Testing (IST) Program Bases Document, Rev. 0000: 1/7/12, 5/26/12, 6/7/12, 8/16/12, 10/11/11, 11/1/12, 11/6/12, 11/22/12, 11/23/11, 12/17/12, 12/18/12, 12/19/12, 1/8/2008, 3/30/2008, 6/22/2008, 9/14/2008, 12/17/2008, 3/2/2009, 5/31/2009, 8/16/2009, 11/15/2009, 2/14/2010, 5/16/2010, 8/21/2010
2-47E225-100, Harsh Environmental Data, Rev. 02
2-47E225-103, Harsh Environmental Data EL 519.0, Rev. 03
2-SI-4.5.C.1(3), RHRSW pump and header operability and flow test from :
3-47E225-100, Harsh Environmental Data, Rev. 07
3-47E225-103, Harsh Environmental Data EL 519.0, Rev. 09
3-OI-71, Reactor Core Isolation Cooling System, Rev. 51
Apparent Cause Evaluation (ACE) for PER 652786
Apparent Cause Evaluation for PER 635729
DCN 40472 disc material change request form for 0-FCV-67-48
DWG 0-47W391-9, Fire Protection – 10CFR50, App. R Penetration, Internal Conduit Fire Seals, Rev. 3
DWG 1-47E858-1 Unit 1 Flow Diagram RHRSW System
DWG 1-47E858-1-ISI ASME Section XI RHR Service Water System Code Class Boundaries
DWG 1-47E859-1 Unit 0 and 1 Flow Diagram EECW System
DWG 1-47E859-1-I Unit 0 and 1 Flow Diagram EECW System
DWG 2-45B891-3, Conduit & Grounding Cable – Conduit Seal Misc Matl Method, Rev. 1
DWG 2-47E858-1 Unit 2 Flow Diagram RHRSW System
DWG 2-47E859-1 Unit 2 Flow Diagram EECW System
DWG 3-47E858-1 Unit 3 Flow Diagram RHRSW System
DWG 3-47E859-1 Unit 3 Flow Diagram EECW System
Functionality Evaluation for PER 635729
Functionality Evaluation for PER 635729 RHRSW Header low pressure alarm received during 2-SI-4.5.C.1(2) EECW Pump Operation
Historical RHRSW Pump performance surveillances related to Intake pump house ground water leakage (0-SI-3.1.3)

Letter from Electric Power Research Institute (EPRI) dated December 3, 2012 to Mark Cook at TVA's Power Service Shops
Maintenance Strategy for BFN-0-FCV-067-0048 dated 2/20/2013
NPG-SPP-09.18.2 Equipment Reliability Classification, Rev. 0001
NPG-SPP-09.2, Equipment Environmental Qualification (EQ) Program, Rev. 01
NRC Event Notification (EN) # 48622, Part 21 Report – Belden Wire used on Environmentally Qualified Form-Wound Motors Not Fully Qualified
PER 623106, Potential Non-conservative Assumptions in Calculations for Leakage in RHRSW Pump Room Sump
PER 635729
PER 652786, Documentation of Belden EPDM motor leads used in Environmentally Qualified motor windings
PER 659017
PER 666222, Issue Associated with RHRSW and Flooding
PER 666225 Operability Review for Groundwater leaks surrounding the intake pump house area.
PER 666247
PER 669508, Temporary Alteration Identified in A RHRSW Pump Room
PER 671475, Seals for Electrical Penetrations
PER 671475, Seals for Electrical Penetrations
PER 675339
PER 687912, RCIC System Operation during Plant Shutdown
PER 940127
PER 957689
Prompt Determination of Operability (PDO) for Per 652786
Prompt Determination of Operability for PER 666222
Prompt Determination of Operability for PER 687912, dated 2/27/13
Verification and Validation Appendix R Manual Actions for 0-SSI-2-4
WO 114327893, Work Order to Address Non-Conformance Documented in PER 666222
WO 45361414
WO-94-020448-000
WO-97-011400-000
WO-97-011425-000
0-TI-230, Predictive Maintenance Program, Rev 25
0-TI-230V, Vibration Program, Rev 12
0-TI-403, Determination of Common Cause Failure for Emergency Diesel Generators, Rev 1
3-SR-3.8.1.1(3D), Diesel Generator 3D Monthly Operability Test, Rev 47, completed 12/24/12
Design Criteria BFN-50-7082, Standby Diesel Generator
FSAR Section 8.5, Standby AC Power Supply and Distribution, BFN-24
NPG-SPP-9.18, Integrated Equipment Reliability Program, Rev 4
NPG-SPP-9.18.2, Equipment Reliability Classification, Rev 1
NPG-SPP-9.18.3, Equipment Reliability Program Component Strategy Development and Implementation Process, Rev 2
NPG-SPP-6.2, Preventive Maintenance, Rev 4
NPG-SPP-6.2.1, Condition Based Maintenance Program, Rev 0
PER 164475, Diesel Generator Blower PM
PER 369956, BFN DG Blower Bearings
PER 488208, Closure Review of PER 369956

PER 665217, DG 3D Generator Fan has Possible Bad Bearing
TVA BFN Component Maintenance Strategy Template: Emergency Diesel Generator ER Strategy Update, Rev 0, dated 1/21/2013
TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan (NQAP) (Quality Assurance Program Description), Rev 27
Unit 3 Technical Specification and Basis 3.8.1 AC Sources - Operating, Amendment 266
Calculation MDQ000992013000171, Calculation of EECW / RHRSW Flow Distribution in the Absence of the RHRSW Pump D1 Crosstie to EECW Valve, BFN-0-FCV-067-0048, During Appendix R Fire Scenarios, Rev. 0

Section 1R18: Plant Modifications
0-SI-4.8.A.1-1(a) Liquid Effluent Batch Release (Other than plant radwaste tanks)
0-SI SI-4.8.A.1-1 Liquid Effluent Permit
Temporary System Drawing – Frac tank release path
NPG-SPP-09.3, Plant Modifications and Engineering Change Control, Rev. 13
NPG-SPP-06.9.3, Post-Modification Testing, Rev. 4
WO 114303359, BFN-0-MVOP-067-0048, Support MMG in T-DCN 40472A/PIC 70944, Rev. 2
WO 114304858, Replace Teledyne Actuator with SMB-000
T-DCN 40472A, Replace Obsolete Crane Butterfly Valve
ECI-0-000-MOV001, Maintenance for Limitorque Motor Operated Valve, Rev. 49
MCI-0-000-ACT002, Maintenance of Limitorque Actuator Model SMB-000, Rev. 33

Section 1R19: Post-Maintenance Testing
0-SI-3.2.10.C, Verification of Remote Position Indicators for Emergency Equipment Cooling Water System Valves, Rev. 0
0-SR-3.3.3.2.1(67), Backup Control Panel Testing, Rev. 8
1-SR-3.3.1.2.5 & 6, Source Range Monitor Functional Test with Reactor Mode Switch Not in Run Position, Rev. 2
1-SR-3.3.1.2.4, Source Range Monitor System Count Rate and Signal to Noise Ratio Check, Rev. 7
3-SR-3.5.1.6(CS II), Core Spray Flow Rate Loop II
SII-0-XX-92-054, IRM/SRM Testing and Temporary Protection Maintenance Instruction, Rev. 9
Design Criteria BFN-50-7075, Core Spray System
ECI-0-000-BKR008, Rev. 94, Testing and Troubleshooting of Molded Case Circuit Breakers and Motor Starter Overload Relays, Rev. 94
ECI-0-000-MOV002, Limitorque Motor Operated Valves Electrical Adjustments, Rev. 25
EPI-0-000-MOV001, Rev. 57, Electrical Preventive Maintenance for Limitorque Motor Operated Valves, Rev. 57
FSAR Section 6.0, Emergency Core Cooling Systems, BFN-24
NPG-SPP-06.9.3, Post-Modification Testing, Rev. 3
Technical Specifications and Bases 3.5.1, ECCS - Operating
WO 111615754, CS Fan B Bearing Modification
WO 112808386, B EDG PMT following bearing replacement
WO 113881247, CS Fan B Bearing Lube
WO 113977958, CS Fan B Breaker PM
WO 113977981, CS II Outboard Discharge Valve PM
WO 113978090, Annual PM CS Fan BW
WO 113978108, CS II Outboard Suction Valve PM

WO 114283473, Replace Valve 0-FCV-067-0048
WO 114303359, BFN-0-MVOP-067-0048, Support MMG in T-DCN 40472A/PIC 70944, Rev. 2
WO 114304858, Actuator replacement
WO 114317762, 0-SR-3.3.3.2.1(67), Backup Control Panel Testing
WO 114317832, 0-SI-3.2.10.C, Verification of Position Indicators
WO 114320973, 0-SI-3.2.10.C, Verification of Position Indicators
WO 114516351, SRM Functional Test
WO 114497817, SRM "C" is Spiking Up and Downscale on Unit 1, Troubleshoot and Repair
WO 114527085, SRM System Count Rate and Signal to Noise Ratio Check

Section 1R20: Refueling and Other Outage Activities
Procedure 2-SR-3.4.9.1() – Reactor Heatup and Cooldown Rate Monitoring
SR 665523, Inspect CCW Tunnel During Unit 1 Outage
PER 687732, Automatic SCRAM due to RPS Actuation
PER 698870, U1 Manual Reactor SCRAM due to Degrading Condenser Vacuum
Unit 3 Reactor Scram Report dated 2/25/2013
Plant Operations Review Committee Meeting Minutes, No. 8874, dated 2/27/2013
Event Number 48782, Automatic Scram Due to a Turbine Trip From a Loss of Condenser Vacuum
Event Number 48829, Manual Reactor Scram Due to Lowering Condenser Vacuum

Section 1R22: Surveillance Testing
Conduct of Testing Records for samples on Batch Releases conducted at 0700 on 1/7/13, 0700 on 1/8/13, and 1430 on 1/10/13
Design Criteria BFN-50-7075, Core Spray System
Drawing 0-45E644-1
Drawing 2-45E765-7
FSAR Section 6.0, Emergency Core Cooling Systems, BFN-24
NPG-SPP-6.9.2, Surveillance Test Program, Rev. 1
NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants
PER 369800-122
PER 566825
PER 673497, Step Improperly Performed during 3-SR-3.5.1.6(CS II) – NRC Identified
PER 673501, 3-SR-3.5.1.6(CS II) Flow Rate Test Delayed Unnecessarily
Procedure 0-SI-3.2.33 (CW A), 1 & 2 Control Bay CHW Pump A Augmented Inservice Test
Procedure 0-SI-4.5.C.1(D3), RHRSW Pump D3 IST Group A Quarterly Pump Test
Procedure 0-SI-4.8.A.1-1 Liquid Effluent Permit
Procedure 0-SI-4.8.A.1-1(a) Liquid Effluent Batch Release (other than plant radwaste tanks)
Procedure 0-SR-3.8.1.9(B), B Emergency Diesel Load Acceptance Test
Procedure 0-TI-362
Procedure 0-TI-444
Procedure 1/2/3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Revs. 2, 7, and 6 respectively
Procedure 1-SI-4.6.B.1-4, Reactor Coolant Chemistry, Rev. 19
Procedure CI-403, Reactor Building Sampling, Rev. 77
SR 694588
Technical Specifications and Bases 3.5.1, ECCS - Operating
Tennessee Valley Authority Offsite Dose Collection Manual (ODCM)

WO 113800615, Perform U3 CS II Flow Rate
WO 114308011
Work Order 113890467, Perform PM P2407AJ – perform 2-SR-3.6.1.3.10(D)
0-TI-412, Work Permits, Rev. 28

Section 4OA1: Performance Indicator (PI) Verification
1/2/3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Revs. 2, 7, and 6 respectively
1-SI-4.6.B.1-4, Reactor Coolant Chemistry, Rev. 19
CI-138, Reporting NEI Indicators, Rev. 3
FSAR Section 14.6.5, Main Steam Line Break Accident, BFN-24
NPG-SPP-2.2, Performance Indicator Program, Rev. 4
NRC ROP Digital City Website PIs as of 3/01/2013 for Browns Ferry Units 1/2/3 Reactor Coolant System Activity and Reactor Coolant System Leakage
PER 346011, Incorrect Max RCS Identified Leakage Reporting
PER 521506, Develop Departmental Desktop Guide for RCS Leakage PI
PER 694496, Incorrect Max RCS Leakage Reporting

Section 4OA2: Identification and Resolution of Problems
Technical Specification (TS) 5.5.2 Primary Coolant Sources Outside Containment
0-TI-578 Minimizing Primary Coolant Sources Outside Containment
TS 5.5.2 leakage tracking
Calculation R14 920727 107 Control Room and Offsite Doses due to a Loss of Coolant Accident (LOCA) Rev 21
Calculation R14 981211 106 Parameters Used in Dose Analysis Rev 6
Calculation EWR11PROG999053 Current Emergency Core Cooling System (ECCS) Leakage
Calculation Leakage Outside Containment Program dated 3/21/2013
Amendment number 251 to License No. DPR-33
Amendment number 290 to License No. DPR-52
Amendment number 249 to License No. DPR-68
PER 317464 TS 5.5.2 addresses primary sources outside containment
PER 348278 Discrepancy between FSAR and 0-TI-19
PER 699268 TS 5.5.2 Program Document Improvements Identified
Surveillance 1-SI-3-3-8-A dated 11/25/2012
Surveillance 2-SI-3-3-8-A dated 8/24/2012
Surveillance 3-SI-3-3-8-A dated 5/5/2012
Surveillance 1-SI-3-3-8-B dated 11/16/2012
Surveillance 2-SI-3-3-8-B dated 2/26/2011
Surveillance 3-SI-3-3-8-B dated 4/10/2012
Surveillance 1-SI-3-3-8-C dated 11/14/2010
Surveillance 2-SI-3-3-8-C dated 4/2/2011
Surveillance 3-SI-3-3-8-C dated 5/6/2012

Section 4OA3: Event Follow-up
PER 539040, BFN-1-FCV-073-0018, HPCI TURB STOP VLV failed to trip during 1-SR-3.5.1.7 HPCI flow rate
Past Operability Evaluation for PER 539040, Rev. 1
MCI-0-073-VLV001, HPCI FCV 73-18 Disassembly, Inspection, Rework and Assembly, Rev. 19

RCA 539040, PER 539040, BFN-1-FCV-073-0018, HPCI Turbine Stop Valve Failed To Trip During 1-SR-3.5.1.7 HPCI Flow Rate, Rev. 3
LER 05000259/2012-010-00, Primary Containment Isolation Valve Inoperable for Longer than Allowed by the Technical Specifications
1-SR-3.6.1.3.8(4), Instrument Line Excess Flow Check Valve Operability Test, Rev. 4
MCI-0-000-CKV002, Maintenance of Marotta Instrument Line Check Valves, Rev. 15
PER 646600, Failed Acceptance Criteria during 1-SR-3.6.1.3.8(4)
WO 113225804, Instrument Line Excess Flow Check Valve Operability Test
WO 112971469, Contingency Repair Marotta Valve as Required
UFSAR Section 14.6, Analysis of Design Basis Accidents – Uprated, Amendment 24
RCA PER 646600, Failed Acceptance Criteria for 1-ECKV-068-0065B, Rev. 0
DWG 1-47E610-68-1, Mechanical Control Diagram Reactor Recirculation System, Rev. 23
DWG 1-47E817-1, Flow Diagram Nuclear Boiler, Rev. 46
PER 562343, Excessive Number of Unit 3 Unplanned Scrams
PER 558437, Manual scram during Unit 3 reactor startup
PER 439393, IRM 3C has enough noise chatter that it is spiking and causing a half-scram
PER 381140, Maintenance Rule Plant Level Performance Criteria for unplanned ESF actuations
PER 144272, Unit 3 Reactor Scram while transferring power to 4kV Unit Board 3B
PER 164325, IRM 1D range switch causing half-scram
PER 135889, 3C IRM half scram
PER 362057, IRM 2G is spiking
PER 375372, Unit 2 ‘G’ IRM erratic
PER 373365, Full Scram due to SDV high water level Unit 3
PER 234151, Full scram on June 9, 2010 due to IRM 2C and 2F spiking
PER 338613, 3B IRM drawer high volts connector J7 needs to be replaced
Dataware History for Unit 3 IRM signals from April 8 to April 13, 2013
Dataware History for Unit 3 IRM signals from April 17 to April 26, 2013
NRC Event Notification # 47955
LER 50-296/2008-001, Unanticipated Auto-Start of the Emergency Diesel Generators
LER 50-296/2011-002, Reactor Scram Due to Scram Discharge Volume High Water Level
SII-0-XX-92-054, IRM/SRM Testing and Temporary Protection Maintenance Instruction, Rev. 07
BFN-50-7092, General Design Criteria Document, Neutron Monitoring System, Rev. 8
LTAM BFN-11-0067, Separate IRM/SRM Pre-Amps into Separate J-Boxes
CDE 1041, Cause Determination Evaluation for MR ESF actuations
MREP meeting minutes from July 14, 2011
MREP meeting minutes from July 28, 2011
MREP meeting minutes from August 11, 2011
PER 402414 Downgrade request from B level apparent cause to C level document actions
3-45N3650-2, Unit 3 Wiring Diagrams Unit Control Boards, Panel 9-12, Rev. 01
3-45E703-3, Unit 3 Wiring Diagram, Battery Board 3, RPS PWR Sys, Single Line, Rev. 24
WO 113394203, SRM, Source Range Monitor, Channel A
2-SR-3.3.6.1.3(3DFT), HPCI Steam Line Space High Temperature Functional Test, Rev. 0
2-SR-3.3.6.1.3(3D), HPCI Steam Line Space High Temperature Calibration, Rev. 8
SII-0-TS-00-320, EGS/Fenwal Environmentally Qualified Temperature Switch Assembly and Repair, Rev. 8
2-AOI-64-2B, Group 4 High Pressure Coolant Injection Isolation, Rev. 16
PER 80014, Wire Degradation Found
PER 596706, During Performance of 2-SR-3.3.6.1.3(3DFT) a HPCI Isolation Occurred

RCE Report for PER 596706, HPCI Isolation During 2-SR-3.3.6.1.3(3DFT), Rev. 1
PER 614099, OE Preventable PER
0-TI-230, Predictive Maintenance Program, Rev 25
0-TI-230V, Vibration Program, Rev 12
0-TI-403, Determination of Common Cause Failure for Emergency Diesel Generators, Rev 1
3-SR-3.8.1.1(3D), Diesel Generator 3D Monthly Operability Test, Rev 47, completed 12/24/12
CRP-ENG-8-9, TVA NPG Emergency Diesel System, dated 9/4/8 to 11/7/8
Design Criteria BFN-50-7082, Standby Diesel Generator
FSAR Section 8.5, Standby AC Power Supply and Distribution, BFN-24
NPG-SPP-9.18, Integrated Equipment Reliability Program, Rev 4
NPG-SPP-9.18.2, Equipment Reliability Classification, Rev 1
NPG-SPP-9.18.3, Equipment Reliability Program Component Strategy Development and Implementation Process, Rev 2
NPG-SPP-6.2, Preventive Maintenance, Rev 4
NPG-SPP-6.2.1, Condition Based Maintenance Program, Rev 0
PER 164475, Diesel Generator Blower PM
PER 369956, BFN DG Blower Bearings
PER 488208, Closure Review of PER 369956
PER 665217, DG 3D Generator Fan has Possible Bad Bearing
PER 667866, Vibration Data for 3D DG Auxiliaries Not Downloaded
PER 675339, Diesel Generator 3A Generator Blower Bearing Lubricant is Degraded
PER 675952, Diesel Generator 3B Generator Blower Bearing Lubricant is Degraded
TVA BFN Component Maintenance Strategy Template: Emergency Diesel Generator ER Strategy Update, Rev 0, dated 1/21/2013
TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan (NQAP) (Quality Assurance Program Description), Rev 27
Unit 3 Technical Specification and Basis 3.8.1 AC Sources - Operating, Amendment 266

LIST OF ACRONYMS

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