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

November 13, 2012

EA-12-199

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 05000259/2012004, 05000260/2012004, AND 05000296/2012004 AND NOTICE OF ENFORCEMENT DISCRETION

Dear Mr. Shea:

On September 30, 2012, 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 October 11, 18 and November 6, 2012, 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.

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

In addition, 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

The enclosed report also documents three non-compliances for which the NRC is exercising enforcement discretion in accordance with Section 9.1 of the NRC Enforcement Policy, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)." The non-compliances are associated with your implementation of the requirements and standards of 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." The non-compliances were identified by the licensee, and are violations of NRC requirements. The inspectors have screened the violations and determined that they warrant enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues, and Section 11.05(b) of IMC 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/William Jones RA for/

Richard P. Croteau, Director
Division of Reactor Projects

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

Enclosure: NRC Integrated Inspection Report 05000259/2012004, 05000260/2012004, and 05000296/2012004

cc w/encl. (See page 3)

The enclosed report also documents three non-compliances for which the NRC is exercising enforcement discretion in accordance with Section 9.1 of the NRC Enforcement Policy, “Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48).” The non-compliances are associated with your implementation of the requirements and standards of 10 CFR 50, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.” The non-compliances were identified by the licensee, and are violations of NRC requirements. The inspectors have screened the violations and determined that they warrant enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues, and Section 11.05(b) of IMC 0305.

In accordance with 10 CFR 2.390 of the NRC’s “Rules of Practice,” a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC’s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/William Jones RA for/

Richard P. Croteau, Director
Division of Reactor Projects

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

Enclosure: NRC Integrated Inspection Report 05000259/2012004, 05000260/2012004, and 05000296/2012004

cc w/encl. (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE
ADAMS: ☐ Yes ACCESSION NUMBER:__________ X SUNSI REVIEW COMPLETE

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS
SIGNATURE	Via email	Via email	Via email	Via email	Via email	Via email	Via email
NAME	DDumbacher	CStancil	PNiebaum	LPressley	TStephen	DJones	JMongomery
DATE	11/08/2012	11/08/2012	11/09/2012	11/08/2012	11/07/2012	11/07/2012	11/08/2012
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
OFFICE	RII:EICS	RII:DRP	RII:DRP	RII:DRP	RII:DRP		
SIGNATURE	Via email	EFG /RA for/	EFG /RA/		WBJ /RA for/		
NAME	JBaptist	CKontz	EGuthrie	WJones	RCroteau		
DATE	11/09/2012	11/13/2012	11/13/2012		11/13/2012		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\DRPII\RPB6\BROWNS FERRY\REPORTS\2012\004\BROWNS FERRY 2012 004.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, 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 Richard P. Croteau dated November 13, 2012

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2012004, 05000260/2012004, AND 05000296/2012004 AND NOTICE OF ENFORCEMENT DISCRETION

Distribution w/encl:
C. Evans, RII
L. Douglas, RII
OE Mail
RIDSNRRDIRS
PUBLIC
RidsNrrPMBrownsFerry Resource

NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2012004, 05000260/2012004, 05000296/2012004
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:	July 1, 2012, through September 30, 2012
Inspectors:	D. Dumbacher, Senior Resident Inspector C. Stancil, Resident Inspector P. Niebaum, Resident Inspector L. Pressley, Resident Inspector T. Stephen, Reactor Inspector C. Kontz, Senior Project Engineer (1R12) D. Jones, Senior Reactor Inspector (4OA5) J. Montgomery, Reactor Inspector (4OA5)
Approved by:	Eugene F. Guthrie, Chief Special Project, Browns Ferry Division of Reactor Projects

SUMMARY

IR 05000259/2012004, 05000260/2012004, 05000296/2012004; 07/01/2012 – 09/30/2012; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Identification and Resolution of Problems, and Event Follow-up.

The report covered a three month period of inspection by the resident inspectors, and three regional inspectors. Two self-revealing findings and one non-cited violations (NCV) were identified. The significance of inspection findings are indicated by their color (Green, White, Yellow, and Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross-Cutting Areas". All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy. 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.

NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding (FIN) was identified for the licensee's failure to provide an adequate design review of vendor calculations as required by TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan which resulted in the 3A Unit Station Service Transformer (USST) differential current protection relay trip settings being incorrectly set. The licensee reset and adequately tested the function of the relay. The licensee has evaluated vendor-provided modifications for similar protective relays and plans to revise the design review process to provide increased licensee accountability and specificity of reviews for vendor designs. The licensee entered this issue into their corrective action program as problem evaluation report (PER) 555573.

  This finding was determined to be more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability. Specifically, the failure to provide an adequate design review of vendor calculations directly contributed to a reactor scram of Unit 3. The significance of the finding was evaluated using Phase 1 of the Significance Determination Process (SDP) in accordance with Inspection Manual Chapter 0609 Attachment 4 and Appendix A and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting aspect of Complete Documentation in the Resources component of the Human Performance area, because the licensee failed to ensure procedure NEDP-5, Design Document Reviews was consistent with TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan [H.2.(c)]. (Section 4OA3.2)

- Green. A self-revealing finding (FIN) was identified for the licensee's failure to adequately test a Unit 3 main turbine generator current transformer (CT) as required by TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan which resulted in the improper wiring of the CT. The licensee switched the CT leads to correct the input to the main transformer relay, adequately tested all other new Unit 3 relays, implemented a transition plan to incorporate the protective relay group into the nuclear organization, and planned post startup monitoring for the Unit 1 and 2 digital differential protective relays. The licensee entered this issue into their corrective action program as PER 558183.

  This finding was determined to be more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability. Specifically, the failure to adequately test a Unit 3 main turbine generator CT directly contributed to a reactor scram of Unit 3. The significance of the finding was evaluated using Phase 1 of the Significance Determination Process (SDP) in accordance with Inspection Manual Chapter 0609 Attachment 4 and was determined to be of very low safety significance (Green) because it did not contribute to both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The cause of this finding was directly related to the cross-cutting aspect of Supervisory and Management Oversight in the Work Practices component of the Human Performance area, because the supervisors failed to ensure proper procedure quality, procedure usage, worker qualification, and proper work preparation associated with the protective relay group's work activities such that nuclear safety was supported [H.4.(c)]. (Section 4OA3.4)

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a non-cited violation (NCV) of 10 CFR 50.54(q)(2) for the licensee's failure to follow and maintain an emergency plan that meets the requirements of emergency planning standard 10 CFR 50.47(b)(4). Specifically, due to a plant modification, the licensee failed to maintain configuration control of seismic instrumentation necessary for the declaration of emergency events from August 17 to August 31, 2012. Completion of installation of the power and instrumentation logic signal to the control room annunciators on August 31, 2012, restored compliance with the emergency plan requirements. The licensee entered this issue into their corrective action program as PER 610625.

  This finding was determined to be more than minor because it was associated with the Emergency Response Organization (ERO) Performance Attribute of the Emergency Preparedness Cornerstone and affected the cornerstone objective of ensuring the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, one Alert and one Notification of Unusual Event Emergency Action Level (EAL) initiating condition would have been rendered ineffective such that a seismic event may not have been appropriately declared. The significance of this finding was evaluated in accordance with the IMC 0609, Appendix B, "Emergency Preparedness

Significance Determination Process," and was determined to be of very low safety significance because an ineffective or degraded EAL scheme that affects Alert declarations was categorized as a Green violation. The cause of this finding was directly related to the cross cutting aspect of Documents, Procedures and Component Labeling in the Resources component of the Human Performance area. Specifically, a lack of complete, accurate and up-to-date design documentation resulted in a loss of configuration control and degradation of information necessary to classify a seismic event. [H.2(c)], (Section 4OA2.4)

Violations of very low safety significance or Severity Level IV 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 essentially full Rated Thermal Power (RTP) except for two planned downpowers of 20 percent for rod pattern adjustments on September 9 and September 21, 2012. Each occurrence lasted for one day.

Unit 2 operated at essentially full RTP most of the inspection period except for four planned downpowers and one unplanned downpower. On August 14, 2012, an unscheduled one-day reduction to 95 percent power was made to perform alternate Main Steam Isolation Valve (MSIV) leakage path testing on 2B Reactor Feedwater Pump. On September 8, 2012, a one-day planned downpower to 70 percent power was made to repair 2B2 recirculation pump variable frequency drive cooling pump. On September 23, 2012, a one-day planned downpower to 80 percent was made to support control rod sequence exchange. On September 27, 2012, a one-day planned downpower to 95 percent was made to remove 2C1 and 2C2 High Pressure Feedwater Heaters from service to Furmanite a small feedwater leak. The Furmanite repair could not be completed and another power reduction and increase back to 100 percent power was made on September 29, 2012 to unisolate the heater. The small feedwater leak still existed at the end of the quarter.

Unit 3 operated at essentially full RTP most of the inspection period except for one planned downpower and one unplanned downpower. On August 26, 2012, a planned downpower to 50 percent power was made to repair pipe supports and a steam leak on the elbow of the moisture separator condenser drain lines. On September 17, 2012, an unplanned downpower to 80 percent power as a result of a loss of the DC control power to the 3EC diesel during governor modification testing. The loss of the control power caused a reduction of main condenser vacuum and the operators initiated a power reduction to 80 percent RTP.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Evaluate Readiness to Cope with External Flooding

a. Inspection Scope

The Inspectors reviewed licensee flood protection barriers and procedures for coping with external flooding. The inspection covered the FSAR and related flood analysis documents to identify those areas that can be affected by external flooding and seasonal susceptibilities such as floods caused by hurricanes, heavy rains and flash floods. The review covered design flood level documentation and corrective actions for safety-related areas. The inspectors conducted a walkdown of the Unit 1, 2, and 3 reactor building entrance flood door. The inspectors reviewed documentation and observed preventive maintenance (PM) activities for the door. Specific focus addressed: Sealing of equipment below the flood line, such as electrical conduits; Sealing of equipment floor plugs, holes or penetrations in floors and walls between flood areas; and Adequacy of watertight doors between flood areas. This activity constitutes one External Flood Protection Sample.

- August 7, 2012, Common Unit Reactor Building Flood Gate as part of Temporary Instruction (TI) -187, Flooding Walkdowns

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors conducted 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, Updated Final Safety Analysis Report (UFSAR), system operating procedures, and Technical Specifications 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 inspection samples.

- Unit 1 High Pressure Coolant Injection (HPCI) System as part of TI-188, Seismic Walkdowns, August 8, 2012
- Unit 2 Reactor Core Isolation Cooling (RCIC) System with High Pressure Coolant Injection (HPCI) out-of-service, August 17, 2012
- Unit 3 Core Spray (CS) System as part of T1-188, Seismic Walkdowns, August 8, 2012.

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 D Emergency Diesel Generator (EDG), using the applicable P&ID flow diagrams, 0-45E724-4, 0-15E500-1, 0-47E610-1, 0-47E861-8, 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 Updated Final Safety

Analysis Report (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, and any previous PERs that could affect system alignment and operability. This constitutes one complete walkdown inspection sample.

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/FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. 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.

- Unit 2 Control Building, EL 593' including auxiliary instrument and communications rooms (Fire Area 16)
- Unit 3 Control Building, EL 593' including computer and auxiliary instrument rooms (Fire Area 16)
- Unit 1/Unit 2/Unit 3 Control Building, EL 606' including cable spreading rooms (Fire Area 16)
- Unit 1/Unit 2/Unit 3 Control Building, EL 617' (Fire Area 16)
- Radiological Waste Building, EL 565' and 580' (Fire Area 26)

b. Findings

No findings were identified.

1R06 Flood Protection Measures

.1 Areas Susceptible to Internal Flooding

a. Inspection Scope

The inspectors performed a review of the Unit 1 main battery and battery 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 Moderate Energy Line Break Flood Evaluation Report for Unit 1-Extended Power Uprate. Furthermore, the inspectors reviewed the Browns Ferry Nuclear Plant Probabilistic Safety Assessment Initiating Event Notebook, Initiating Event Frequencies, for licensee commitments. The inspectors performed walkdowns of risk-significant areas, susceptible systems and equipment, including the Unit 1 main battery rooms, Unit 2 main battery rooms and associated board 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 sampling of the licensee's corrective action documents with respect to flood-related items to verify that problems were being identified and corrected. Furthermore, the inspectors reviewed selected completed preventive maintenance procedures, work orders, and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents. This activity constituted one Internal Flood Protection inspection sample.

b. Findings

No findings were identified.

.2 Cables Located in Underground Bunkers/Manholes

a. Inspection Scope

The inspectors conducted an inspection of underground bunkers/manholes subject to flooding that contain cables whose failure could disable risk-significant equipment. The inspectors performed walkdowns of the following underground areas containing safety-related and/or risk significant cables; Hand-Hole (HH) 15 and HH-26 located on the east-side of the reactor building. These walkdowns were conducted to verify that safety-related and/or risk-significant cables were not submerged in water, or water damaged; all cables and/or splices appeared intact; and the proper condition of associated cable tray support structures. As applicable, the inspectors verified proper operation of installed dewatering devices (i.e., sump pumps) and level switches to ensure that affected cables would not become submerged. Where dewatering devices were not installed, the inspectors ensured that drainage was provided and was functioning properly.

Furthermore, the inspectors reviewed past preventative maintenance activities performed by the licensee to inspect plant manholes, valve pits, and cable tunnels; and check operability of applicable sump pumps. This activity constituted one Underground Manhole flooding inspection sample.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification and Performance

.1 Requalification Activities

a. Inspection Scope

On July 23, 2012, the inspectors observed an as-found licensed operator requalification for an operating crew according to Unit 2 Simulator Exercise Guide OPL173S315, Turbine Trip, Anticipated Transient without Scram.

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

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

The inspectors attended the post-examination critique to assess the effectiveness of the licensee evaluators, and to verify that licensee-identified issues were comparable to issues identified by the inspector. The inspectors also reviewed simulator physical fidelity (i.e., the degree of similarity between the simulator and the reference plant control room, such as physical location of panels, equipment, instruments, controls, labels, and related form and function). This activity constituted one Observation of Requalification Activity 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 specific structures, systems and components (SSC) within the scope of the Maintenance Rule (MR) (10CFR50.65) with regard to some or all of the following attributes, 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. 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-03.1, 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 three Maintenance Effectiveness inspection samples.

- Unit 3 Start-Up Operational Justification for In-Service and Motor Operated Valve Testing Degraded or Nonconforming Conditions
- Unit Common Emergency Equipment Cooling Water (EECW) Header Strainers
- Preventive Maintenance Controls for Components In-service Beyond Designated Service Life

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 (RMAs) 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.

- July 3, 2012, Planned modification work on B Emergency Diesel Generator
- July 10, 2012, Planned work on the common B1 RHRSW pump and header, Unit 3 3A CRD pump for motor replacement, Unit 3 Core Spray Division 2 for heavy lift above system, Unit 3 3EA LPCI motor-generator set preventive maintenance, environmental factor adjustments for Unit 1 for 1B Main Transformer oil leak
- August 9, 2012, Planned maintenance on Unit 1 RHR Loop II, A2 RHRSW pumps, A EECW Strainer, with D Emergency Diesel Generator out of service
- September 11, 2012, Planned maintenance (12 year Preventative Maintenance (PM)) and modification work on 3C Emergency Diesel Generator
- September 27, 2012, Planned maintenance (12 year PM) and governor replacement on 3B Emergency Diesel Generator with C1 RHRSW pump out of service

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, 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 eight Operability Evaluation inspection samples.

- A Standby Gas Treatment Iodine Removal Efficiency Test (PER 574215)
- C Standby Gas Treatment relative humidity breaker tripped (PER 554624)
- Control Malfunction of the B Emergency Diesel Generator Operational Mode Select Switch from the Unit 1/2 Control room (PER 597415)
- Unit 3 Reactor Vessel Head Deformation Due to Foreign Object (PER 538810)
- Safety Related 250V DC Unit Batteries Remote Ammeter Cables Not Adequately Protected (PER 452185)
- U1 RHRSW and RHR Loop II Loss of Safety Function Issues (PER 594986)
- D Emergency Equipment Cooling Water (EECW) Strainer Clearance Nonconformance with Design Requirements (PER 605866)
- 3B Emergency Diesel Generator Lube Oil Usage (PER 424092)

b. Findings

No findings were identified.

1R18 Plant Modifications

a. Inspection Scope

The inspectors reviewed the modification listed below to verify regulatory requirements were met, along with procedures, as applicable, 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 screenings and evaluations and compared each against the UFSAR and Technical Specifications to verify that the modifications did not affect operability or availability of the affected systems. Furthermore, the inspectors walked down each 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 constituted one Plant Modification inspection sample.

- Design Change Notice (DCN) 69532, 3C Emergency Diesel Generator mechanical governor replacement with a new 2301A electronic load sharing and speed control governor

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors witnessed and reviewed the 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 six Post Maintenance Test inspection samples.

- Unit 3: Post Modification Testing Instruction (PMTI) for 3A USST Differential Current Protection Relay per PMTI-61731-004
- Unit 1: Post-Modification Test for B Emergency Diesel Generator Lube Oil system pump modifications per WO 113239540 and PMTI 69454 STG002
- Unit 3: Post-Modification Test of Main Turbine Generator Current Transformer to New Main Transformer per Work Orders 112346955 and 112350792, and procedure TOM-FTM-6-INXF-002, Testing Instrument Transformers
- Unit 3: Post Modification Testing Instruction (PMTI) for Generator Voltage Regulator Replacement per PMTI-70095-001 and PMTI-70095-002
- Unit 3: Post Modification Testing Instruction (PMTI) for 3C Emergency Diesel Generator governor control upgrade PMTI-69532-STG007
- Unit Common A Standby Gas Treatment Fan High Vibrations per WOs 113883328 and 113883379; and maintenance procedures EPI-0-000-MOT001, Motor Bearing Lubrication, and MPI-0-000-BLT001, Belt Drive Maintenance

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 Technical 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 six inspection samples, two in-service and four routine surveillance tests.

In-Service Tests:

- September 5, 2012, 2-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure
- September 29, 2012, 2-SR-3.5.1.6(CS I), Core Spray Flow Rate Loop I

Routine Surveillance Tests:

- July 5, 2012, 2-SR-3.5.3.3, RCIC System Rated Flow at Normal Operating Pressure
- July 20, 2012, 1-SR- 3.3.1.1.2, APRM Output Signal Adjustment
- August 13, 2012, 3-SR-3.8.1.1(3DR), Diesel Generator 3D Operability Test (App R), which also satisfies requirements of the Appendix R Safe Shutdown Program to test and load the Diesel from 4kV Shutdown Board 3ED
- September 17, 2012, 3-SR-3.8.1.1, Diesel Generator 3C

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) drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures on August 22, 2012, to identify any 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, and certain Emergency Response Facilities to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure, and other applicable Emergency Plan Implementing Procedures. The inspectors also attended the post-drill critique to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying weaknesses. This inspection activity satisfied one inspection sample for the Drill Evaluation of emergency preparedness.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Mitigating Systems

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following performance indicators (PIs), including procedure NPG-SPP-02.2, Performance Indicator Program. The inspectors examined the licensee's PI data for High Pressure Coolant Injection for the second quarter of 2011 through the second quarter of 2012. The inspectors examined the licensee's PI data for the Cooling Water, Emergency Power and Residual Heat Removal (RHR) PIs listed below for the third quarter 2011 through the second quarter 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 go over licensee records to verify that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors also used the Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to ensure that industry reporting guidelines were appropriately applied.

- Unit 1 Mitigating Systems Performance Index – High Pressure Coolant Injection
- Unit 2 Mitigating Systems Performance Index – High Pressure Coolant Injection
- Unit 3 Mitigating Systems Performance Index – High Pressure Coolant Injection
- Unit 1 Mitigating Systems Performance Index – Residual Heat Removal
- Unit 2 Mitigating Systems Performance Index – Residual Heat Removal
- Unit 3 Mitigating Systems Performance Index – Residual Heat Removal
- Unit 1 Mitigating Systems Performance Index - Cooling Water System (RHRSW/EECW)
- Unit 2 Mitigating Systems Performance Index - Cooling Water System (RHRSW/EECW)
- Unit 3 Mitigating Systems Performance Index - Cooling Water System (RHRSW/EECW)
- Unit 1 Mitigating Systems Performance Index - Emergency AC Power

- Unit 2 Mitigating Systems Performance Index - Emergency AC Power
- Unit 3 Mitigating Systems Performance Index - Emergency AC Power

This constituted 12 total samples for the four above MSPI Systems.

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: Corrective Actions Associated with Licensee Response to Operating Experience – Switchgear Fire at Fort Calhoun

a. Inspection Scope

The inspectors reviewed the Browns Ferry response to significant industry operating experience (OE) related to the Fort Calhoun Station switchgear fire which occurred in June 2011. The cause of the OE fire was associated with an inadequate connection of cradle cluster finger contacts of modified Nuclear Logistics Incorporated (NLI) Square D Masterpact 480 volt breakers with switchgear bus stabs. Browns Ferry was replacing these low voltage breakers due to aging and obsolescence. The inspectors focused on the ongoing accelerated replacement of approximately 42 breakers with the new NLI breakers during the Unit 3 2012 refueling outage. The inspectors specifically questioned the applicability of the industry OE related to the Fort Calhoun Station switchgear fire. The licensee initiated PERs 523757 and 546288 and documented corrective actions which the inspectors reviewed to assess the effectiveness and adequacy of the licensee’s response. The inspectors also interviewed engineering and maintenance personnel to understand similarities between the OE and the licensee’s breaker replacements. The inspectors observed inspections and breaker replacements of several 480 volt switchgears. This activity constituted one Follow-up of Selected Issues inspection sample.

b. Findings and Observations

No findings were identified. The inspectors discussed the following observations with the licensee. In general, the inspectors determined that the licensee’s evaluation of the OE was thorough and consistent with the licensee’s processes. The corrective actions implemented and scheduled were considered reasonable to address the cause of the

OE related fire. The inspectors observed weaknesses associated with the licensee's evaluation of the 480 volt switchgear back plane (bus bar) area. Current inspection and preventive maintenance, historical documentation, and drawings did not exist for the bus bars. The licensee was working with the vendor to resolve these issues under the above noted PERs.

.3 Annual Follow-up of Selected Issues: Emergency Equipment Cooling Water (EECW) Strainer Excessive Barrel and Seal Ring Clearances

a. Inspection Scope

The inspectors reviewed the specific causal analyses and corrective actions for licensee initiated PER 243132, EECW Diesel Generator (DG) Functional Failure; PER 254463, Degraded Conditions Not Identified and Evaluated, Based On Reliability, for the EECW & Residual Heat Removal (RHR) Systems (Low EECW Flow Recurring Issues); PER 381569, 3D DG Inoperable due to Low EECW Flow; PER 605866, D EECW Strainer Nonconformance with Design Requirements; and the EECW Pumps Maintenance Rule (a)(1) Plan, Revision 2, EECW Strainers. The inspectors interviewed engineering personnel to assess the effectiveness and adequacy of the licensee's efforts to correct the recurring fouling issues noted on various safety-related Core Spray and RHR room coolers and Emergency Diesel Generator heat exchangers. The inspectors focused their review on corrective actions taken to address the conditions identified, including subsequent operability evaluations; the extent of condition analyses; and the prioritization of the corrective actions. Additionally, the inspectors evaluated these elements against regulatory requirements and the licensee's CAP. This review constituted one Problem Identification and Resolution (IP 71152) annual inspection sample.

b. Findings and Observations

No findings were identified.

.4 Focused Annual Sample Review - Operator Workarounds

a. Inspection Scope

The inspectors conducted a review of an existing Operator Workaround (OWA) to verify that the licensee was identifying OWAs at an appropriate threshold, entering them into the corrective action program, establishing adequate compensatory measures, prioritizing resolution of the problem, and implementing appropriate corrective actions in a timely manner commensurate with its safety significance. On September 15, 2012, the inspectors examined active workaround LCOTR 0-052-OWA-2012-0169 associated with a modification of the seismic monitoring system and reviewed it against the guidance in BFN-ODM-4.16, Operator Workarounds/ Burdens/Challenges. The inspectors also discussed this OWA in detail with on shift operators to assess their familiarity with the degraded conditions and knowledge of required compensatory actions. Furthermore, the inspector walked down this selected OWA, and verified the ongoing performance, and/or feasibility of, the required actions. Lastly, for the selected OWA, the inspectors reviewed the applicable corrective action document, including the associated functional evaluation and corrective action plans (both interim and long term). This activity constituted one Operator Workaround inspection sample.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green), and an associated NCV of 10 CFR 50.54(q)(2) for the licensee's failure to follow and maintain an emergency plan that met the requirements of emergency planning standard 10 CFR 50.47(b)(4). Specifically, the licensee failed to maintain configuration control of seismic instrumentation necessary for the declaration of emergency events.

Description: On November 8, 2011, the Browns Ferry staff performed design change reviews to approve new seismic monitoring equipment per procedure, NPG-SPP-09.4 "10 CFR 50.59 Evaluation of Changes, Tests, and Experiments." Work on the three-train Seismic Monitor System commenced on August 13, 2012, to support modification DCN 69995A. On August 17, 2012, the entire system was made non-functional due to the old scratch pads and all the cabling from the seismic sensors to the control room annunciators being disconnected as part of the modification work. The operators declared the entire system inoperable but functional. On September 13, 2012, the NRC resident inspector questioned the control room operating crew whether the annunciators were functional. This crew believed the annunciators were not functional. However, station engineering and emergency preparedness staff stated that the system was available and functional. Initial compensatory actions were implemented as operator work around guidance on September 15, 2012. After NRC questioning the system availability again on September 19, the licensee determined that the system had been out-of-service to support maintenance from August 17 – 30, 2012. The initial operator work around guidance was incomplete and lacked clarity. Without the control room annunciators, no established guidance described an alternative method to evaluate the emergency action level (Alert, 7-1-A) ground motion acceleration threshold of 1.0g for the Operating Basis Earthquake described in the Final Safety Analysis Report.

On September 20, 2012, the operator workaround guidance was revised to facilitate the awareness of the state of the modification, clarify the use of newly worded annunciator window tiles, and provide the operators a, specified but not previously provided, Mercalli Intensity scale. These changes were needed to ensure the operators ability to assess and report a seismic related Alert condition. This and other corrective actions were captured in PER 610625.

Analysis: The licensee's failure to maintain configuration control of seismic instrumentation necessary for the declaration of emergency events in accordance with the site emergency plan was a performance deficiency. This finding was determined to be of more than minor significance because it was associated with the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone and affected the cornerstone objective of ensuring the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, one Alert Emergency Action Level initiating condition would have been rendered ineffective such that a seismic event may not have been appropriately declared. The significance of this finding was evaluated in accordance with the IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," section 5.4, 10 CFR 50.47(b)(4), Emergency Classification System Table 5.4.1, issued 2/2412, which states that an emergency action level rendered ineffective such that any Alert could not be declared or declared in a degraded manner is a Green finding.

The cause of this finding was directly related to the cross cutting aspect of Documents, Procedures and Component Labeling in the Resources component of the Human Performance area. Specifically, a lack of complete, accurate and up-to-date design documentation resulted in a loss of configuration control and degradation of information necessary to classify a seismic event. (H.2(c)).

Enforcement: Title 10 CFR 50.54(q) required, in part, that licensee's follow and maintain an emergency plan that meets the requirements of emergency planning standard 10 CFR 50.47(b)(4). Title 10 CFR 50.47(b)(4) required, in part, that a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures." Contrary to the above, the licensee failed to maintain an emergency plan that met the requirements of 10 CFR 50.47(b)(4). Specifically, from August 17 – 30, 2012, the seismic monitor system was out-of-service rendering the Browns Ferry facility emergency plan incapable of providing information needed for determinations of minimum initial offsite response measures related to the Alert emergency classification action level 7-1-A. On August 30, 2012, the licensee restored power to the annunciators which restored compliance with the emergency plan. Because this violation was of very low safety significance and because it had been entered into the licensee's corrective action program as PER 610625, the violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 5000259, 260, 296/2012004-01), Loss of Seismic Monitoring Capability.

4OA3 Follow-up of Events

.1 (Closed) Licensee Event Report (LER) 05000259, 260, and 296/2012-005-00 and 05000259, 260, and 296/2012-005-01, Combustible Materials not in Compliance with the 20-Foot Exclusion Zone Requirements

a. Inspection Scope

During December 2011, TVA staff conducted an extent of condition inspection/walkdown of exclusion zone "red floor" areas and identified transient combustible material in the Units 1, 2 and 3 reactor buildings. The red floor exclusion zones were established for compliance with the 20-foot separation requirements of 10 CFR 50 Appendix R Section III.G.2.b. The exclusion zones that contained the transient combustibles separate fire zones 1-1 from fire zone 1-2 and the two trains of the RHR low pressure injection (LPCI) outboard isolation and flow control valves. On March 28, 2012, TVA determined that the transient combustibles materials found in the exclusion zone were not in compliance with

10 CFR 50 Appendix R Section III.G.2.b and the NRC approved exemptions for this fire area. On March 29, 2012, TVA made an 8-hour report to the NRC in accordance with 10 CFR 50.72(b)(3)(ii)(B). See Event Notification #47787. The licensee entered this issue into their corrective action program (CAP) as PERs 529001 and 558964. The inspectors reviewed the details surrounding this event and licensee's apparent cause evaluation, and verified the adequacy of corrective actions. Additional documents reviewed are documented in the Attachment. This LER is closed.

b. Findings

The enforcement aspects of this finding are discussed in Section 4OA7.

.2 (Closed) Licensee Event Report (LER) 50-296/2012-003-00, 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

On May 22, 2012, Unit 3 automatically scrammed from approximately 19 percent RTP 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 initial follow-up of this event by inspectors was documented in Section 4OA3.1 of IR 05000296/2012003. The inspectors reviewed the applicable LER that was issued on July 23, 2012, and it's associated PER 555573, which included the root cause analysis (RCA) and corrective actions. The licensee concluded that the direct cause of the Unit 3 scram was the incorrect design calculation settings on the differential relay.

b. Findings

This LER is considered closed with one finding identified.

Introduction: A Green self-revealing finding (FIN) was identified for the licensee's failure to provide an adequate design review of second party engineering calculations. Consequently the 3A Unit Station Service Transformer (USST) differential current protection relay trip settings were incorrectly set resulting in a Unit 3 automatic scram on May 22, 2012.

Description: 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. Unit 3 was recovering from a refueling outage with the reactor at power and the turbine not synchronized to the electrical grid. With the unit off-line, offsite electrical power to the safety-related 4KV shutdown boards flows from the 500KV switchyard, through the 500KV/22KV main transformer and the 22KV/4KV 3B USST to the 3A and 3B 4KV Unit Boards. During the activity to transfer 3C Unit Board (non-safety-related loads) from the alternate to the normal power supply, the differential current protection relay on the 3A USST actuated resulting in de-energization of the main transformer and a loss of power to the Unit 3 safety related 4KV shutdown boards. As designed, the Reactor Protection System de-energized which caused a reactor scram.

The inspectors responded to the scram event to ensure the plant conditions were stable. The inspectors reviewed the licensee's root cause analysis (PER 555573) and determined the conclusions and corrective actions were appropriate. The direct cause of the trip was an incorrect design calculation setting for a new Unit 3 USST differential current protection digital relay which was replaced during the refueling outage as part of the 3A USST replacement modification. Vendor engineering design calculation EDN324320100007, Unit 3 USST 3A Transformer Differential Relay Settings, incorrectly assumed a standard USST transformer phase shift and input an incorrect phase shift into the differential current protection relay. The licensee's quality assurance program as described in TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan, Section 7.2.3.B, Design Analysis, required the licensee to ensure the suitability of application of equipment, and processes essential to the function of a structure, system, or component be reviewed to ensure that functional requirements were met. The licensee determined that the design review implementing procedures NEDP-5, Design Document Reviews and DS-M18.1.3, Engineering Procurement and Vendor Technical Quality, inadequately defined the requirements for reviews of vendor engineering calculations to meet the requirements of TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan. Specifically, DS-M18.1.3, Engineering Procurement and Vendor Technical Quality, incorrectly stated that licensee engineering should not assume the responsibility for detailed checking of the vendor information. Therefore, the licensee did not perform the technical review necessary to identify the incorrect differential current protection relay setting The licensee reviewed other similar vendor-provided relay modifications and plans to revise the design review process and procedures as immediate corrective action,.

<u>Analysis</u>: The inspectors determined that the licensee's failure to adequately review vendor design calculation EDN324320100007 to identify and correct an erroneous transformer phase shift of the differential current protection relay was a performance deficiency. Specifically, the licensee failed to meet the requirements of procedure TVA-NQA-PLN89-A, Section 7.2.3.B, Design Analysis, to ensure the suitability of application of equipment, and processes essential to the function of a structure, system, or component be reviewed to ensure that functional requirements were met. This finding was determined to be more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability. Specifically, the failure to provide an adequate design review of vendor calculations directly contributed to a reactor scram of Unit 3. The significance of the finding was evaluated using Phase 1 of the SDP in accordance with Inspection Manual Chapter 0609 Attachment 4, Initial Characterization of Findings, and Appendix A, SDP for Findings At-Power, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting aspect of Complete Documentation in the Resources component of the Human Performance area, because the licensee failed to ensure procedure NEDP-5, Design Document Reviews was consistent with TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan. (H.2.c).

Enforcement: This finding does not involve enforcement action because no violation of a regulatory requirement was identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as FIN 05000296/2012004-02, Automatic Reactor Scram Due to Inadequate Design Review of Relay Setting.

.3 (Closed) Licensee Event Report (LER) 05000260/2012-002-00, High Pressure Coolant Injection System Rendered Inoperable Due to an Inoperable Primary Containment Isolation Valve

a. Inspection Scope

The inspectors reviewed the LER for potential performance deficiencies and/or violations of regulatory requirements. The LER was associated with a steam leak on the Unit 2 High Pressure Coolant Injection (HPCI) system 73-81, HPCI warm up bypass valve, a Primary Containment Isolation Valve (PCIV). The valve was declared inoperable and the required actions associated with Technical Specification 3.6.1.3 were taken to isolate the upstream PCIV, valve 73-2, HPCI steam inboard isolation valve, which rendered the HPCI system inoperable. The inspectors reviewed 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 566687. Additional documents reviewed are listed in the Attachment. This LER is closed.

b. Findings

The enforcement aspects of this finding are discussed in Section 4OA7.

.4 (Closed) Licensee Event Report (LER) 50-296/2012-005-00, Automatic Reactor Scram Due to an Actuation of a Main Transformer Differential Relay

a. Inspection Scope

On May 29, 2012, Unit 3 automatically scrammed from approximately 75 percent RTP when the premature actuation of the main transformer differential over-current protection relay caused a main generator load rejection signal. The initial follow-up of this event by inspectors was documented in Section 4OA3.3 of IR 05000296/2012003. The inspectors reviewed the applicable LER that was issued on July 30, 2012, and it's associated PER 558183, which included the root cause analysis (RCA) and corrective actions. The licensee concluded that the direct cause of the Unit 3 scram was the reversed polarity current transformer (CT) which provided a false signal to the differential relay.

b. Findings

This LER is considered closed with one finding identified.

Introduction: A Green self-revealing finding (FIN) was identified for the licensee's failure to adequately test a Unit 3 main turbine generator current transformer (CT). Consequently the main transformer differential over-current protection relay was prematurely actuated by a reversed polarity CT resulting in a Unit 3 automatic scram on May 29, 2012.

Description: On May 29, 2012, Unit 3 automatically scrammed from approximately 75 percent RTP when the premature actuation of the main transformer differential over-current protection relay caused a main generator load reject signal. The inspectors responded to the scram event to ensure plant conditions were stable. The inspectors reviewed the licensee's root cause analysis (PER 558183) and determined the conclusions and corrective actions were appropriate. The licensee determined that the main transformer relay 387T actuated because a manufacturing defect in a new main turbine generator current transformer (CT) caused an incorrect excitation input to relay 387T. Both were new components installed during the recent Unit 3 refueling outage. The manufacturing defect was the reversed polarity (improperly wired) of one of 36 new CTs, 3-CTR-242-5B2L-1.

The licensee determined that the CT pre-installation bench test for polarity was inadequate, thereby, failing to identify the CT with the reversed polarity. The licensee also determined the root cause of the reactor scram to be inadequate management oversight and accountability for the protective relay group transition to the licensee's nuclear organization and work performed by the relay group. The licensee's quality assurance program as described in TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan, Section 9.4, Test Control, in part, required tests to confirm that component replacements produce expected results and do not reduce operations safety. The technician that performed the test was not qualified because of an inadequate training program for the protective relay group. The procedure to test the CT, TOM-FTM-6-INXF-002, Testing Instrument Transformers, a carryover from the relay group's non-nuclear transmission division, did not specify level of use, critical steps, second party verification, or acceptance criteria, and was not approved for use by the nuclear organization. The approved work order, the test scoping document, and the post-modification test control form all specified the inadequate CT test procedure. The unqualified technician worked alone and did not use the test procedure for guidance. He relied on skill of the craft and a sketch produced by an experienced technician for equipment setup. The polarity test required the visual verification of a positive DC "kick" of 2 to 5 percent of voltmeter scale, which was challenging, even for experienced technicians. The licensee determined that a post startup monitoring plan for the energized main transformer had not been prescribed to verify CT inputs to the relays during power ascension. Earlier identification of the defective CT phasing may have provided the operators with an opportunity to remove the main turbine generator offline in a controlled manner.

The licensee switched the CT leads to correct the input to the main transformer relay, adequately tested all other new Unit 3 relays, implemented a transition plan to incorporate the protective relay group into the nuclear organization, and planned post startup monitoring for the Unit 1 and 2 digital differential protective relays. The licensee concluded more dependable test methodologies, digital equipment, and use of larger power sources were needed to increase the validity of the tests.

Analysis: The inspectors determined that the licensee's failure to adequately test the Unit 3 main turbine generator current transformer (CT) was a performance deficiency. Specifically, the licensee failed to meet the requirements of TVA procedure TVA-NQA-PLN89-A, Section 9.4, Test Control, which required tests to confirm that the replaced components produced expected results and did not reduce operations safety. The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability. Specifically, the failure to adequately test the Unit 3 main turbine generator CT directly contributed to a reactor scram of Unit 3. The significance of the finding was evaluated using Phase 1 of the SDP in accordance with Inspection Manual Chapter 0609 Attachment 4, Initial Characterization of Findings, and Appendix A, SDP for Findings At-Power, and was determined to be of very low safety significance (Green) because it did not contribute to both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The cause of this finding was directly related to the cross-cutting aspect of Supervisory and Management Oversight in the Work Practices component of the Human Performance area, because the supervisors failed to ensure proper procedure quality, procedure usage, worker qualification, and proper work preparation associated with the protective relay group's work activities such that nuclear safety was supported (H.4.c).

Enforcement: This finding does not involve enforcement action because no violation of a regulatory requirement was identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as FIN 05000296/2012004-03, Automatic Reactor Scram Due to Inadequate Testing of Current Transformer.

.5 (Closed) LER 05000259, 260, and 296/2011-002-01, Loss of Safety Function (SDC) Resulting from Loss of Power from C EDG Due to Oil Leak

a. Inspection Scope

The inspectors reviewed Revision 1 of the LER dated March 21, 2012. This revised LER was submitted to provide the results of the licensee's completed investigation and evaluation of past operability. The original LER 50-259/2011-002-00 dated June 27, 2011, and applicable PER 362395, including cause determination and corrective action plans, were reviewed by the inspectors and documented in Section 4OA3.1 of NRC IR 05000259/2011004. No violation of NRC requirements was initially identified.

On April 27, 2011, severe weather in the Tennessee Valley service area caused a reactor scram of all three Browns Ferry units. On April 28, 2011, control room personnel performed an emergency shutdown of the Unit 1 and 2 common 'C' Emergency Diesel Generator (EDG) due to a hydraulic oil leak on tubing for the EDG governor that was causing voltage and frequency fluctuations. The shutdown of the 'C' EDG resulted in the loss of shutdown cooling to Units 1 and 2. The licensee determined the immediate cause was a leaking one-eighth of an inch threaded brass fitting on the governor-to-governor booster pump hydraulic oil tubing. The licensee determined the root cause to be less than adequate vendor design of the Unit 1 and 2 common 'C' EDG governor hydraulic oil tubing to compensate for vibration loading. As documented in the revised LER, the licensee concluded that the 'C' EDG would not have fulfilled its 7-day mission time from April 1 – 30, 2011.

The inspectors reviewed the LER revision and verified that the supplemental information provided in the 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 licensee performance deficiency was identified by the inspectors.

b. Findings

No findings were identified. This LER is closed.

.6 (Closed) Licensee Event Report (LER) 50-260, 296/2012-001-00, Browns Ferry Nuclear Plant, Units 2 and 3, Inappropriate LOCA Modeling of Core Spray for Limiting LOCA Event with Manual Actuation of Automatic Depressurization System

a. Inspection Scope

The inspectors reviewed LER 05000260, 296/2012-001-00 dated June 12, 2012, and applicable PERs 372764, 527811, 539468 and 515520, including cause determinations and corrective actions. This LER was submitted to report the failure of the licensee to accurately maintain design control regarding single-failure assumptions for the Emergency Core Cooling System (ECCS) and the failure to perform sufficiently bounding analyses to ensure that the calculated maximum fuel element cladding temperature of 2200 degrees Fahrenheit would not be exceeded in the event of a small break loss of coolant accident in accordance with 10 CFR 50.46. Unresolved Item (URI) 05000259, 260, 296/2011003-03, Use of Inappropriately Qualified Methods to Evaluate Emergency Core Cooling during Accident Mitigation, was previously documented in Section 4OA5.3 of NRC Inspection Report (IR) 05000259, 260, 296/2011003, and closed in NRC IR 05000259, 260, 296/2012002. As a result of the URI closure, two violations of NRC requirements were identified: NCV 05000260, 296/2012002-03, Failure to Ensure ECCS Design Calculation Does Not Exceed Maximum Clad Temperature, and NCV 05000260, 296/2012002-04, Repeated Failure to Report ECCS Analyses Methodology Change or Errors.

The licensee determined the root cause to be inappropriate reviews for design document control associated with ECCS availability in a LOCA scenario. Corrective actions included revised LOCA analyses, interim thermal limit penalties, design changes to remove single-failure vulnerability of the Automatic Depressurization System, and programmatic revisions for document reviews and regulatory interface. During the causal determination for PER 515220, 10 CFR 50.46 30 Day Reporting Requirement Not Met, the licensee further determined that annual 10 CFR 50.46(a)(3)(ii) reports for years 2007, 2008, and 2009 had not been submitted as a result of the licensee's misinterpretation of the regulatory intent as documented in the Statement of Considerations for 10CFR 50.46 (53 FR 36001, September 16,1988). The licensee failed to continue to report the change reported on November 7, 2006, until a revised evaluation model or revised evaluation correcting minor errors was approved by the NRC staff. No revised evaluation was completed or approved during this timeframe. The licensee documented this issue in their CAP as PER 533361. This is a minor performance deficiency because the required reports, had they been submitted, would have carried through the 2006 reported change and would not have resulted in increased regulatory response. Although this issue should be corrected, this failure to comply with 10 CFR 50.46 constitutes a minor violation that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.

The inspectors verified that the supplemental information provided in the LER was complete and accurate and that the information was not of a significant nature to warrant any change to the URI findings.

b. Findings

The enforcement aspects of this LER were discussed in NRC IR 05000259, 260, 296/2012002. No additional findings were identified. This LER is closed.

.7 (Closed) Licensee Event Report (LER) 05000259/2012-006-00, and 05000259/2011-006-01, High Pressure Coolant Injection System Turbine Failed to Trip Using the Manual Trip Push Button.

a. Inspection Scope

The inspectors reviewed LER 05000259/2012-006-00 dated June 18, 2012, and the revised LER 05000259/2012-006-01 dated July 18, 2012. Inspectors reviewed PER's 539040 and 377771 related to this event. The initial LER stated verification of correct installation on Units 2 and 3 had been performed and that a supplement was forthcoming following further analysis. The revised LER provided additional information including; root cause, contributing factors, further 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 particularly reactor core isolation cooling (RCIC). 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; see Licensee Identified Violations Section 4OA7. No additional findings were identified regarding the original or revised LER. These LERs are closed.

.8 (Closed) Licensee Event Reports (LER) 05000296/2012-002-00, Main Steam Isolation Valves Leakage in Excess of Technical Specification Requirements

a. Inspection Scope

The inspectors reviewed LER 05000296/2012-002-00 dated June 06, 2012, and applicable PER 553052, including cause determinations and corrective actions. This LER was submitted to report the failure of surveillance procedure 3-SR-3.6.1.3.1 0, Primary Containment Local Leak Rate Test Main Steam Line B the 3B Outboard Main Steam Isolation Valve (MSIV) on April 7, 2012. Specifically the valve failed to meet the minimum technical specification leak rate limit of 100 standard cubic feet per hour (scfh). The as-found leak rate was 781.09 scfh. The MSIV 3-FCV-001-0027 did not have recent maintenance performed that could have contributed to the packing leak. The licensee determined, over time, that the packing preload was lost due to packing relaxation and that the current valve packing program did not give adequate guidance on maintaining the valve packing. The inspectors reviewed the event and determined that the licensee entered and exited the appropriate TS action statement following discovery and no violation of TS occurred.

b. Findings

No findings were identified. This LER is closed.

.9 (Closed) Licensee Event Report (LER) 05000259/2012-001-00, Unanalyzed Conditions Discovered During NFPA 805 Transition Review

a. Inspection Scope

The inspectors reviewed LER 2012-001-00 that documented a deficiency in the fire protection program. The LER documented several unanalyzed conditions associated with multiple spurious operation (MSO) of equipment during postulated fires that were identified during the licensee's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed in-plant walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee's compensatory measures and corrective actions.

b. Findings

Introduction. The licensee identified a non-compliance with 10 CFR Part 50, Appendix R, Section III.G.2 for the licensee's failure to protect one of the redundant trains of cables and equipment, located in the same FA, needed to achieve post-fire SSD from fire damage. Specifically, the licensee failed to use one of the means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage.

Description. On April 5, 2012, the licensee submitted LER 2012-001-00, which documented several unanalyzed conditions associated with MSO of equipment during postulated fires. The LER stated that fire induced hot shorts could adversely affect the reactor pressure instrument loops, safety relief valve overpressure logic, automatic depressurization logic, RHR test return valves, drywell spray valves, suppression pool spray valves, and main steam isolation valves (MSIV). Additionally, the LER documented the potential de-energization of a 4kV shutdown board. The licensee determined that the deficiencies existed because their current Appendix R analyses did not consider MSO of equipment during postulated fires to be credible. This LER was applicable to Units 1, 2 and Unit 3.

The licensee entered this issue into their corrective action program as PER 229734, PER 259787, and PER 424389 and implemented compensatory actions in the form of fire watches for Units 1, 2, and 3. The licensee discovered the MSO vulnerabilities during the review of their Appendix R design bases as part of the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

Analysis. Failure to protect one train of cables and equipment necessary to achieve post-fire SSD from fire damage for fire areas designated in the Fire Protection Report (FPR) as meeting Appendix R, Section III.G.2, was a performance deficiency. This finding was more than minor because it was associated with the reactor safety mitigating system cornerstone attribute of protection against external events (i.e., fire). Specifically, failure to protect safe shutdown cables and equipment from fire damage negatively affected the reactor safety mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because this issue relates to fire protection and this non-compliance was identified as a part of the site's transition to NFPA 805, this issue is being dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy. In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less-than-Red (i.e. ΔCDF less than 1E-4/year). The inspectors also performed walkdowns to verify key assumptions were applicable.

Based on the ignition frequency of fire sources in the affected areas, combined with the probability of non-suppression for those fire scenarios, inspectors determined that the significance of this non-compliance was less-than-Red. The inspectors also noted that the values in the licensee's quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red).

The inspectors determined that no cross cutting aspect was applicable to this performance deficiency because this finding was not indicative of current licensee performance.

Enforcement. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G.

10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

- separation of cables and equipment by a fire barrier having a 3-hour rating,
- separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area, and
- enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on April 5, 2012, the licensee identified the failure to protect equipment that was required to mitigate fire events. The licensee determined that fire damage could prevent operation or cause mal-operation of the reactor pressure instrument loops, safety relief valve overpressure logic, automatic depressurization logic, RHR test return valves, drywell spray valves, suppression pool spray valves MSIVs, and a 4kV shutdown board due to hot shorts, open circuits, or shorts to ground. This condition has existed since initial plant startup for Units 1, 2 and 3. The licensee entered this issue into the corrective action program (PERs 229734, 259787 and 424389) and implemented compensatory actions in the form of fire watches for Units 1, 2, and 3.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for these issues in accordance with the NRC Enforcement Policy, Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" and Inspection Manual Chapter 0305. Specifically, these issues were identified and will be addressed during the licensee's transition to NFPA 805, they were entered into the licensee's corrective action program, immediate corrective action and compensatory measures were taken, they were not likely to have been previously identified by routine licensee efforts, they were not willful, and they were not associated with a finding of high safety significance (Red). This LER is closed

.10 (Closed) Licensee Event Report (LER) 05000259/2012-002-00, Fault Propagation During a Postulated Appendix R Event Could Result in an Inability to Close Motor Operator Valves

a. Inspection Scope

The inspectors reviewed LER 2012-002-00 that documented a deficiency in the fire protection program. The LER documented that certain postulated fires could result in fire induced damage of control circuits for motor operated valves. The licensee identified the deficiency during the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee's compensatory measures and corrective actions.

b. Findings

Introduction. The licensee identified a non-compliance with 10 CFR 50, Appendix R, Section III.G.2 for the licensee's failure to protect one of the redundant trains of cables and equipment, located in the same FA, needed to achieve post-fire SSD from fire damage. Specifically, the license failed to provide adequate protection for the control circuits of 24 motor operated valves (MOVs) that are utilized in safe shutdown procedures to mitigate fire events.

Description. On April 5, 2012, the licensee submitted LER 2012-002-00, which described a previously unanalyzed condition concerning fire protection. The LER documents that certain postulated fires could result in fire induced circuit damage of MOV control circuits. It was determined that fire damage could adversely affect the open/close limit switch and/or torque switch functions of the MOV control circuitry. Damaged circuitry could cause the affected actuator's motor to run continuously - until thermal overload devices de-energized the motor. This damage could subject the MOVs to forces that exceed design limits. This LER was applicable to Units 1, 2 and 3.

The licensee entered this issue into their corrective action program as PER 245385 and PER 245386, and implemented compensatory actions in the form of fire watches for Units 1, 2, and 3. PER 245385 identified 24 MOVs that were not adequately protected; these valves included the emergency equipment cooling water (EECW) pump cross-tie valves, EECW RHR heat exchanger outlet valves, and the main steam drain to condenser valves. The licensee discovered the deficiencies during the review of their Appendix R design bases as part of the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

Analysis. Failure to protect one train of cables and equipment necessary to achieve post-fire SSD from fire damage for fire areas designated in the FPR as meeting Appendix R, Section III.G.2, was a performance deficiency. This finding was more than minor because it was associated with the reactor safety mitigating system cornerstone attribute of protection against external events (i.e., fire). Specifically, failure to protect safe shutdown cables and equipment from fire damage negatively affected the reactor safety mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because this issue relates to fire protection and this non-compliance was identified as a part of the site's transition to NFPA 805, this issue is being dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy. In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less-than-Red (i.e. ΔCDF less than 1E-4/year). The inspectors also performed walkdowns to verify key assumptions were applicable.

Based on the ignition frequency of fire sources in the affected areas, combined with the probability of non-suppression for those fire scenarios, inspectors determined that the significance of this non-compliance was less-than-Red. Inspectors also noted that the values in the licensee's quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red).

The team determined that no cross cutting aspect was applicable to this performance deficiency because this finding was not indicative of current licensee performance.

<u>Enforcement</u>. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G.

10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

- separation of cables and equipment by a fire barrier having a 3-hour rating,
- separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area, and
- enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of the means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on April 5, 2012, the licensee identified the failure to protect the control circuitry of 24 MOVs that were required to mitigate fire events. The licensee determined that fire damage could cause mal-operation of the redundant trains of equipment due to hot shorts, open circuits, or shorts to ground. This condition has existed since initial plant startup for Units 1, 2 and 3. The licensee entered this issue into the corrective action program (PERs 245385 and 245386) and implemented compensatory actions in the form of fire watches for Units 1, 2, and 3.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for these issues in accordance with the NRC Enforcement Policy, Section 9.1, “Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)” and Inspection Manual Chapter 0305. Specifically, these issues were identified and will be addressed during the licensee’s transition to NFPA 805, they were entered into the licensee’s corrective action program, immediate corrective action and compensatory measures were taken, they were not likely to have been previously identified by routine licensee efforts, they were not willful, and they were not associated with a finding of high safety significance (Red). This LER is closed

.11 (Closed) Licensee Event Report (LER) 05000259/2012-003-00: Reactor Protection System Circuit Could Potentially Remain Energized During An Appendix R Fire

a. Inspection Scope

The inspectors reviewed LER 2012-003-00 that documented a deficiency in the fire protection program. The LER documented the discovery of a condition where 120V Alternating Current (AC) lighting circuitry was installed in Reactor Protection System (RPS) cabinets without electrical isolation. Due to this lack of isolation between Class 1E and Non-Class 1E circuits, a fire affecting a non-safety related circuit could render safety-related equipment unavailable. The licensee identified the deficiency during the site’s transition to NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.”

The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed in-plant walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee’s compensatory measures and corrective actions.

b. Findings

Introduction. The licensee identified a non-compliance with 10 CFR Part 50, Appendix R, Section III.G.2 for the licensee’s failure to protect one of the redundant trains of cables and equipment, located in the same FA, needed to achieve post-fire SSD from fire damage. Specifically, the licensee failed to provide adequate circuit protection for associated non-safety circuits, and/or to institute protection methods for these circuits in accordance with the separation criteria specified in 10 CFR Part 50, Appendix R, Section III.G.2. These protection methods include the use of spatial separation, passive fire barriers, fire detection, and automatic fire suppression.

Description. On April 9, 2012, the licensee submitted LER 2012-003-00, describing a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shut down the reactor and maintain it in a safe condition. During the review of their Appendix R design bases as part of the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," the licensee discovered that RPS cabinets 9-15 and 9-17 on all three BFN units contained a 120V lighting (utility) circuit feed. This 120V AC lighting circuit provided power to the cabinets' internal lights and duplex receptacles. This LER was applicable to Units 1, 2 and Unit 3.

The RPS cabinets contain separate 120V AC circuits that are required to be de-energized to initiate a scram, in accordance with BFN Safe Shutdown Instructions (SSIs), in the event of a serious fire. Due to the lack of physical separation with the 120V AC lighting circuitry, the RPS circuits could remain energized due to a postulated hot short during a fire, which could prevent the control rods from inserting into the reactor. The RPS cabinets that are affected are located in Auxiliary Instrument Room for each unit. These rooms are located in the control building (Fire Area 16), which is common for all 3 units.

Procedure 0-SSI-16 for FA 16 "Control Bay" contains actions to open the RPS breakers to ensure power is removed from the scram valves. However, this operator manual action (OMA) could be rendered ineffective due to the presence of the 120V AC lighting circuits in the cabinet that could, due to hot short, spuriously energize the RPS bus in non-protected parts of the circuit.

In reference to Appendix R associated circuits, Design Criteria BFN-7200C required that associated circuits that share a common enclosure with an Appendix R required circuit must be protected by the use of a fuse or breaker. Upon discovery that the lighting circuit did not contain any electrical isolation, the licensee entered the condition into the corrective action program. At the time of discovery, compensatory measures, in the form of fire watches, were already in place for the affected FA, due to previously identified non-compliances.

Analysis. Failure to protect one train of cables and equipment necessary to achieve post-fire SSD from fire damage for fire areas designated in the FPR as meeting Appendix R, Section III.G.2, was a performance deficiency. This finding was more than minor because it was associated with the reactor safety mitigating system cornerstone attribute of protection against external events (i.e., fire). Specifically, failure to protect safe shutdown cables and equipment from fire damage negatively affected the reactor safety mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because this issue relates to fire protection and this non-compliance was identified as a part of the site's transition to NFPA 805, this issue is being dispositioned in accordance with Section 9.1, “Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)” of the NRC Enforcement Policy.

In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less than Red (i.e. ΔCDF less than 1E-4/yr.). The inspectors also performed walkdowns to verify key assumptions were applicable.

Based on the ignition frequency of fire sources in the affected areas, combined with the probability of non-suppression for those fire scenarios, inspectors determined that the significance of this non-compliance was less than Red. Inspectors also noted that the values in the licensee’s quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red).

The inspectors determined that this non-compliance did not have a cross-cutting aspect because it did not represent current licensee performance.

<u>Enforcement</u>. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G.

10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

- separation of cables and equipment by a fire barrier having a 3-hour rating,
- separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area, and
- enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of the means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on February 8, 2012, the licensee discovered that RPS cabinets 9-15 and 9-17 on all three BFN units contains 120V lighting (utility) circuit feeds that are not separated or isolated from safety related circuits located in RPS cabinets. This condition could result in a fire in the RPS cabinet causing a hot short that could result in the inability to scram the reactor. This condition has existed since initial plant startup for all 3 units. The licensee entered this issue into the corrective action program

(PER 503304). At the time of discovery, compensatory measures, in the form of fire watches, were already in place for the affected FA, due to previously identified non-compliances.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for these issues in accordance with the NRC Enforcement Policy, Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" and Inspection Manual Chapter 0305. Specifically, these issues were identified and will be addressed during the licensee's transition to NFPA 805, they were entered into the licensee's corrective action program, immediate corrective action and compensatory measures were taken, they were not likely to have been previously identified by routine licensee efforts, they were not willful, and they were not associated with a finding of high safety significance (Red). This LER is closed

4OA5 Other Activities

.1 (Discussed) NRC Temporary Instruction (TI) 2515/187, Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns, and NRC TI 2515/188, Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

Inspectors accompanied the licensee on a sampling basis, during their flooding and seismic walkdowns, to verify that the licensee's walkdown activities used the methodology endorsed by the NRC and initiated identified corrective actions into the licensee corrective action program. These walkdowns are being performed at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 3 of the March 12, 2012, letter requested licensees to perform seismic walkdowns using an NRC-endorsed walkdown methodology. Electric Power Research Institute (EPRI) document 1025286 titled, "Seismic Walkdown Guidance," (ADAMS Accession No. ML12188A031) provided the NRC-endorsed methodology for performing seismic walkdowns to verify that plant features, credited in the current licensing basis (CLB) for seismic events, are available, functional, and properly maintained.

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, are available, functional, and properly maintained.

b. Findings

Findings or violations associated with the flooding and seismic walkdowns, if any, will be documented in future reports.

.2 Operation of an Independent Spent Fuel Storage Installation (ISFSI) (60855)

a. Inspection Scope

Under the guidance of IP 60855.1, the inspectors observed operations involving spent fuel transfer and storage for dry cask campaign number six. Inspectors interviewed personnel and reviewed the licensee's documentation regarding storing spent fuel to verify that these independent spent fuel storage installation (ISFSI) related programs and procedures fulfill the commitments and requirements specified in the Safety Analysis Report (SAR), Certificate of Compliance (CoC), 10 CFR Part 72, the Technical Specifications (TS), any related 10 CFR 72.48 evaluations, and 10 CFR 72.212(b) evaluations for general licensed ISFSIs. In addition, the inspectors observed selected ISFSI related activities and conducted independent evaluation to ensure that the licensee performed spent fuel loading and transport in a safe manner and in compliance with approved procedures. Inspectors performed focused operational reviews on new methodologies concerning forced helium dehydration and supplemental cooling.

The inspectors reviewed twenty, 10 CFR 72.48 Screening Reviews for ISFSI procedure changes and verified that all changes were consistent with the license and CoC, and did not reduce program effectiveness. The inspectors also reviewed the 10 CFR 72.212, Report of Evaluations.

The inspectors performed a focused observation of loading and transfer operations associated with multi-purpose canister (MPC) S/N 0325 and HI-STORM cask S/N 0489. Inspectors attended briefings and observed operations in the field including overall supervisory involvement, coordination, and oversight of ISFSI-related work activities. The inspectors noted that the field supervisor maintained strict control of the work package and continually verified that procedural steps were followed and completed as required. The inspectors reviewed the fuel loading plan for MPC-0325 and verified that the fuel assemblies identified were properly selected and loaded in accordance with characterization documents and approved procedures.

The inspectors verified that selected individuals had received the necessary training in accordance with approved procedures for their ISFSI-related job duties.

The inspectors reviewed work orders, completed procedures, logs, welding records, inspection records, qualification records, and overall guidelines for MPC-325 ISFSI activities. The inspectors determined that the licensee had established, maintained, and implemented adequate control of dry cask processing operations, including loading, transportation, and storage per approved procedures and that technical specification requirements and acceptance criteria as outlined in the Final Safety Analysis Report were followed appropriately. Records of spent fuel stored at the facility were properly maintained. The inspectors verified that changes to the design and operation were appropriately evaluated under 10CFR72.48. The inspectors determined that radiation protection controls were adequately established and implemented.

The inspectors also made direct observations and reviewed selected records to ensure the licensee had identified each fuel assembly placed in the ISFSI, had recorded the parameters and characteristics of each fuel assembly, and had maintained a record of each fuel assembly as a controlled document.

b. Findings and Observations

No findings were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On October 11 and 18 and November 6, 2012, the results of the inspection to Mr. K. Polson, Site Vice President, and other members of the licensee's staff, who acknowledged the findings.

All proprietary information reviewed by the inspectors as part of routine inspection activities were properly controlled, and subsequently returned to the licensee or disposed of appropriately.

4OA7 Licensee-Identified Violations

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

- Technical Specifications 5.4.1.d, Procedures, required that written procedures for the Fire Protection Program shall be established, implemented and maintained. Section 7.2.2.c, Combustible Material Control Procedures of the Fire Protection Plan established that the storage or staging of transient combustibles during modes 1, 2, or 3 would be restricted from within the twenty foot zone of separation visibly marked as "red floors". Contrary to the above, since approximately the year 2000 when the Radiation Protection remote camera system was installed, the licensee failed to adequately control transient combustibles in a "red floor" area in the Units 1, 2 and 3 reactor buildings as required by the Fire Protection Plan. The licensee entered this issue into their CAP as PERs 529001 and 558964. The safety significance of this finding was characterized to be of very low safety significance (Green) in accordance with IMC 0609, Appendix F, because the finding was assigned a low degradation rating and reflected a fire protection program element whose performance was minimally impacted by the inspection finding.

- Technical Specifications 3.6.1.3, Primary Containment Isolation Valves (PCIVs), required that while Unit 2 is in Modes 1, 2, and 3, each PCIV, except reactor building-to-suppression chamber vacuum breakers shall be operable. The Technical

Specification (TS) action statement A.1 required in part that an affected flow path be isolated by use of at least one closed and de-activated automatic valve within four hours. Contrary to the above, between June 7, 2012, and June 13, 2012, while Unit 2 was in Mode 1, the licensee identified a leak coming from the 2-FCV-73-0081, a one-inch HPCI steam warm-up bypass PCIV and action was not taken to isolate the affected penetration flow path within four hours. 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 2-FCV-73-0081 valve was a one-inch valve and would not generally contribute to Large Early Release Frequency (LERF) as discussed in IMC 0609, Appendix H.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Beck, Instrumentation and Controls Engineering Supervisor
J. Boyer, Assistant Director Site Engineering
A. Chapman, ISFSI Project Manager
M. Ellet, Maintenance Rule Coordinator
J. Emens, Licensing Manager
A. Feltman, Emergency Preparedness Manager
V. Furr, PRA Program Manager
K. Groom, Mechanical Design Engineering Supervisor
C. Guey, PRA Senior Manager
W. Hayes, Reactor Engineering Manager
D. Hughes, Operations Manager
D. Kettering, I&C and Electrical Systems Engineering Manager
R. Loggins, Operations Support Superintendent
M. Oliver, Site Licensing
A. Prucha, Operations Department Corrective Actions Group
S. Spears, Electrical Maintenance Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000259,260,296/2012004-01	NCV	Loss of Seismic Monitoring Capability. (Section 4OA2.4)
05000296/2012-004-02	FIN	Automatic Reactor Scram Due to Inadequate Design Review of Relay Setting (Section 4OA3.2)
05000296/2012-004-03	FIN	Automatic Reactor Scram Due to Inadequate Testing of Current Transformer (Section 4OA3.4)

Closed

05000259,260,296/2011-002-01	LER	Loss of Safety Function (SDC) Resulting from Loss of Power from C EDG Due to Oil Leak (Section 4OA3.5)
05000259,260,296/2012-005-00	LER	Combustible Materials not in Compliance with the 20-Foot Exclusion Zone Requirements (Section 4OA3.1)
05000259,260,296/2012-005-01	LER	Combustible Materials not in Compliance with the 20-Foot Exclusion Zone Requirements (Section 4OA3.1)

05000259/2012-006-00	LER	High Pressure Coolant Injection System Turbine Failed to Trip Using the Manual Trip Pushbutton (Section 4OA3.7)
05000259/2012-006-01	LER	High Pressure Coolant Injection System Turbine Failed to Trip Using the Manual Trip Pushbutton (Section 4OA3.7)
05000260, 296/2012-001-00	LER	Browns Ferry Nuclear Plant, Units 2 and 3, Inappropriate LOCA Modeling of Core Spray for Limiting LOCA Event with Manual Actuation of Automatic Depressurization System (Section 4OA3.6)
05000260/2012-002-00	LER	High Pressure Coolant Injection System Rendered Inoperable Due to an Inoperable Primary Containment Isolation Valve (Section 4OA3.3)
05000296/2012-003-00	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 4OA3.2)
05000296/2012-005-00	LER	Automatic Reactor Scram Due to an Actuation of a Main Transformer Differential Relay (Section 4OA3.4)
05000296/2012-002-00	LER	Main Steam Isolation Valves Leakage in Excess of Technical Specification Requirements (Section 4OA3.8)
05000259/2012-001-00	LER	Unanalyzed Conditions Discovered During NFPA 805 Transition Review (Section 4OA3.9)
05000259/2012-002-00	LER	Fault Propagation During a Postulated Appendix R Event Could Result in an Inability to Close Motor Operator Valves (Section 4OA3.10)
05000259/2012-003-00	LER	Reactor Protection System Circuit Could Potentially Remain Energized During An Appendix R Fire Section 4OA3.11)

Discussed

2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

LIST OF DOCUMENTS REVIEWED

Section 1R01: Flood Protection Measures
Drawings 0-44N235 1 through 5
Drawings 0-44N236 1 through 5
Procedure 0-AOI-100-3 R 35, Flood Above Elevation 558’
Procedure MP-0-000-1N001, Flood Gate Arrangement
WO 112545464
WO 113618794

Section 1R04: Equipment Alignment
0-OI-18 EDG Operations
0-OI-82/ATT-1D EDG 1D Valve Lineup Checklist
0-OI-82/ATT-2D EDG 1D Panel Lineup Checklist
0-OI-82/ATT-3D EDG 1D Electrical Lineup Checklist
1-OI-73, High Pressure Coolant Injection System, Rev. 23
2-OI-71, Reactor Core Isolation Cooling, Rev. 65
2-OI-71/ATT-1, Reactor Core Isolation Cooling Valve Lineup Checklist, Rev. 59
2-OI-71/ATT-2, Reactor Core Isolation Cooling Panel Lineup Checklist, Rev. 60
DWG 0-15E500-1 Standby Auxiliary Power System
DWG 0-45E724-4 4160V Shutdown BD D Single Line
DWG 0-47E610-1 EDG 1D Fuel Oil System
DWG 0-47E840-3 EDG 1D Fuel Oil System
DWG 0-47E861-4 EDG 1D Starting Air System
DWG 0-47E861-8 EDG 1D Cooling system and Lubricating Oil System
DWG 1-47E812-1, Flow Diagram High Pressure Coolant Injection System, Rev. 35
PER 596706, During performance of 2-SR-3.3.6.1.3(3DFT), a HPCI Isolation Occurred

Section 1R05: Fire Protection
Fire Protection Report Volume 1, Rev 12
Fire Hazard Analysis Fire Areas 16 and 26, Rev 12
Safe Shutdown Analysis Fire Areas 16 and 26, Rev 12
Fire Protection Impairment Permits 09-1920, 10-2762, 12-3341, 12-3342, 12-3374, 12-3438, 12-3459, 12-3596
Roving Fire watch coverage sheet dated 9/24/2012
SR 615208 Phones not working in cable spreading room
SR 614508 Door BFN-3-DOOR-260-482 needs bottom seal replaced
SR 615819 Light fixtures missing covers in cable spreading room
NPG-SPP-18.4.7 Control of Transient Combustibles
NPG-SPP-18.4.6 Control of Fire Protection Impairments

Section 1R06: Internal Flood Protection Measures

0-15E810-1, Conduit & Grounding Plan, Rev. 1
0-15N810-3, Electrical, Rev. 3
0-15N810-5, Conduit & Grounding Details, Rev. 1
0-15N810-7, Conduit & Grounding Plan, Rev. 1
0-35N800, Conduit & Grounding Floor EL 550.0 Plan, Rev. 3
DCA 50868-014, Electrical
DCN 50868-A, System 67 Cables
NDN-000-999-2007-0031, IF – BFN Probabilistic Risk Assessment – Internal Flooding Analysis calculation
PER 613967, Request Changes to PM 500103184
PER 614083, NRC identified an Issue with Man-Hole 15
PM 500103184, Perform Check of Plant Sump Pumps for Listed Manholes, Handholes, Valve Pits and Tunnels.
WO 112988609, Perform Check of Plant Sump Pumps for Listed Manholes, Handholes, Valve Pits And Tunnels.
WO 113965818, NRC identified an Issue with Man-Hole 15

Section 1R11: Licensed Operator Requalification

Simulator Exercise Guide OPL173S315, Turbine Trip, Anticipated Transient without Scram

Section 1R12: Maintenance Effectiveness

BFN-VTD-K143-0020, Installation and Maintenance Instructions for Kinney Automatic Self-Cleaning Strainers, Model AV Series, Rev. 5
BFN-VTD-K143-0050, Instruction Manual for S. P. Kinney Model AV Series Motorized Automatic Self-Cleaning Strainers, Class 1, Rev. 0
Design Criteria BFN-50-7067, Browns Ferry Nuclear Plant Emergency Equipment Cooling Water System, Rev. 18
Drawing 1-47E859-1, Flow Diagram Emergency Equipment Cooling Water, Rev. 82
FSAR Section 10.10 Emergency Equipment Cooling Water System, BFN-24
PER 243132, EECW DG Functional Failure
PER 254463, Degraded Conditions Not Identified and Evaluated, Based On Reliability, for the EECW & RHR Systems
PER 381569, 3D Diesel Generator Inoperable due to Low EECW Flow
PER 388243, C3 EECW Pump Exceeding Unavailability
PER 391452, A EECW Strainer Failure
PER 605866, D EECW Strainer Nonconformance with Design Requirements
PER 611259, 3A CS Room Cooler Flush per TI-54, required early 11/12
SR 611498, 1A CS Room Cooler Flush per TI-54, required early 10/12
TVA Nuclear Power Group BFN Engineering Support Morning Status, Degraded Conditions/Non-Conforming, dated September 12, 2012
NPG-SPP-06.2, Preventive Maintenance, Rev. 4
NPG-SPP-09.18, Integrated Equipment Reliability Program, Rev. 3
PER 527089, First Time PM’s
Browns Ferry White Paper, BFN Service Life Project, PER 527089, Preliminary Implementation Plan, dated July 6, July 31, and August 30, 2012, Rev. 4

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

PER 593385, Failure to Update Unit 1 PSA/ORAM Risk Color in the POD
NPG-SPP-7.0, Work Management
0-SR-3.8.1.1(TDG Implementation) Temporary Diesel Generator Implementing Surveillance, Rev 13
NPG-SPP-09.11.1, Equipment Out of Service (EOOS) Management, Rev. 4
NPG-SPP-07.2.11, Shutdown Risk Management, Rev. 2
NPG-SPP-07.2.11, Shutdown Risk Management, Rev. 2
NPG-SPP-7.2.11, Shutdown Risk Management, Rev. 2
NPG-SPP-07.1, On Line Work Management, Rev. 5
NPG-SPP-07.2, Outage Management, Rev. 2
NPG-SPP-07.3, Work Activity Risk Management Process, Rev. 7
NPG-SPP-07.3, Work Activity Risk Management Process, Rev. 9
Operator's Risk Report dated 8/9/2012
BFN Operations Logs dated 8/9/2012
NPG-SPP-09.11, Probabilistic Risk Assessment (PRA) Program, Rev. 1
O-TI-367, BFN Equipment to Plant Risk Matrix

Section 1R15: Operability Evaluations

0-OI-82, Standby Diesel Generator System, Rev. 122
3-OI-82, Standby Diesel Generator System, Rev. 108
ASTM D3803-1989, Standard Test Method for Nuclear-Grade Activated Carbon
BFN Unit 1/2/3 Fire Protection Report Volume 1, Fire Protection Plan, Section 9.3.11.g, Fire Rated Assemblies, Rev.12
BFN-50-7065, Standby Gas Treatment System General Design Criteria Document, Rev. 17
BFN-VTD-K143-0020, Installation and Maintenance Instructions for Kinney Automatic Self-Cleaning Strainers, Model AV Series, Rev. 5
BFN-VTD-K143-0050, Instruction Manual for S. P. Kinney Model AV Series Motorized Automatic Self-Cleaning Strainers, Class 1, Rev. 0
Design Criteria BFN-50-7067, Browns Ferry Nuclear Plant Emergency Equipment Cooling Water System, Rev. 18
Design Criteria BFN-50-7200C, Browns Ferry Nuclear Plant 250V DC Power Distribution System
Drawing 1-47E859-1, Flow Diagram Emergency Equipment Cooling Water, Rev. 82
Drawings 1/2/3-45E701/2/3-1, Wiring Diagram Battery Boards 1/2/3 Panels 1-7 Single line, Revs. 63/57/50
Drawings Units 1/2/3-47E605-22, Mechanical Layout Control Panels 1/2/3-9-8, Revs. 1/8/7
FSAR Section 10.10 Emergency Equipment Cooling Water System, BFN-24
FSAR Section 8.6 250V DC Power Supply and Distribution, BFN-20
LER 50-259/2011-010-00, DC Ammeter Cables Not Adequately Isolated
ML042730028, Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term
ML042730028, Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term
MPI-0-082-LUB001, Standby Diesel Engine Oil Addition, Rev. 1
NEDP-22, Operability Determinations and Functional Evaluations, Rev. 12
NEDP-27, Past Operability Evaluations, Rev. 00
NRC Event Notification (EN) 47374

Operating Logs of August 9 & 10, 2012
OWA 0-82-OWA-2011-0196, Lube Oil Usage
OWA 3-82-OWA-2011-0197, Lube Oil Usage
PER 150500, Pinhole Leak in 2A/2C RHRSW Tunnel
PER 303544, ‘C’ SBGT relative humidity heater breaker found tripped
PER 424092, 3B Diesel Generator Lube Oil Usage
PER 445811 QA Finding – Untimely resolution of EDG non-conservative Tech Spec
PER 445825, Past Operability of Lube Oil Consumption Rate
PER 460426, Barrels in Service Building with no "Ownership" documentation
PER 520892, Compensatory Action for PER 424092 - Lube Oil Staging for Diesel Generators
PER 554624, ‘C’ SBGT relative humidity heater breaker tripped while in service
PER 559061, Damaged HEPA filter in ‘A’ SBGT
PER 562958, DG Lube Oil Consumption Degraded Condition
PER 574215, “A’ SBGT failed the initial Iodine removal efficiency test
PER 593009, Through Wall Leak in 2C RHRSW Inlet Piping
PER 593097, Unplanned LCO Entry
PER 594986, U1 RHRSW and RHR Loop II Loss of Safety Function Issues
PER 605866, D EECW Strainer Nonconformance with Design Requirements
PER574934, Past Operability Evaluation for PER 554624
RHRSW Inoperability and Impact to RHRSW Safety Function
Tech Spec Bases 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, Rev. 0
Technical Specifications and Bases 3.7.2 Emergency Equipment Cooling Water System (EECW) and Ultimate Heat Sink (UHS), Amendment 235
Technical Specifications and Bases 3.8.4 DC Sources – Operating, Amendment 235
UFSAR, Section 8.5, Standby AC Power Supply and Distribution, Amendment 24
WO 112753223, Add Oil to the A D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753224, Add Oil to the B D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753225, Add Oil to the C D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753226, Add Oil to the D D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753227, Add Oil to the 3A D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753228, Add Oil to the 3B D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753229, Add Oil to the 3C D/G per Applicable Steps of MPI-0-082-LUB001
WO 112753230, Add Oil to the 3D D/G per Applicable Steps of MPI-0-082-LUB001
WO 112758824, DG A Lube Oil Consumption Rate
WO 112758881, DG B Lube Oil Consumption Rate
WO 112758905, DG C Lube Oil Consumption Rate
WO 112759003, DG D Lube Oil Consumption Rate
WO 112759027, DG 3A Lube Oil Consumption Rate
WO 112759045, DG 3B Lube Oil Consumption Rate
WO 112759059, DG 3C Lube Oil Consumption Rate
WO 112759073, DG 3D Lube Oil Consumption Rate
WO 113150789, Performance of 0-SR-3.6.4.3.2(A) – SBGTS Iodine Removal Efficiency (Train A)

Section 1R18: Plant Modifications
Design Change Notice 69532A

Section 1R19: Post Maintenance Testing

0-SR-3.8.1.1 (B), B EDG Monthly Operability Test, Rev 51
BP-259, NPG TCM Role and Oversight of Supplemental Personnel
DCN 69454-02
DCN Test Scoping Document TSD-61731-004, TSD-E Stage 4 Differential Relays
DS-M18.1.3, Mechanical Design Standard, Engineering Procurement & Vendor Technical Quality
NPG-SPP-6.9.3, Post Modification Testing
Operator log entries for July 2, 2012
PER 484548, Human Performance Shortfalls
PER 505709, Potential Trend in the Human Performance Cross-Cutting Aspect H.2.c
PER 543131, 95003 Fundamental Problem: Technical Rigor
PER 564389, Trend of Vendor Design Technical Rigor and Oversight
PMTI-61731-004, Differential Relays
PMTI-69532-STG007, 3C EDG Governor Control Upgrade
PMTI-6954-STG002, Rev 1
SR 574886
SR 574892
WO 113043566P

Section 1R22: Surveillance Testing

3-SR-3.8.1.1(3DR), Diesel Generator 3D Operability Test (App R), Rev. 25
3-SR-3.8.4.2(DG 3D), Diesel Generator 3D Battery Service Test, Rev. 19
WO 10527175, Diesel Generator 3D Operability Test (App R)
WO 112781748, Diesel Generator 3D Operability Test (App R)
2-SIMI-64B, Primary Containment Systems Scaling and Setpoint Documents, Rev. 42
2-SR-3.5.1.6(CS I), Core Spray Flow Rate Loop I, Rev. 36
Design Criteria BFN-50-7075, Core Spray Cooling System, Rev. 12
Drawing 2.47E814-1, Flow Diagram Core Spray System, Rev. 52
Drawing 2-47E610-64-1, Mechanical Control Diagram Primary Containment System, Rev. 45
Drawing 2-47E779-16, Wiring Diagram 480 V Shutdown Aux Power Schematic Diagram, Rev. 52
FSAR Section 6.4.3, Description - Core Spray System, BFN-24
FSAR Section 6.5.2.4, Performance Analysis - Core Spray System, BFN-24
0-TI-362, In-Service Testing of Pumps and Valves, Rev. 34
SR 616077, Daily Schedule Not Auto Populating Surveillance Template Support Resources
SR 616516, NRC Issues Following Pre-Job Brief of Core Spray Loop I Flow Rate
SR 617458, Electrical Foreman Failure to sign Environmental Qualification Maintenance Work Record for 2-TB-64-8793
Technical Specifications and Bases 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation, Amendment 296
Technical Specifications and Bases 3.5.1, ECCS - Operating, Amendment 294
Technical Specifications and Bases 3.5.2, ECCS - Shutdown, Amendment 294
Technical Requirements Manual and Basis TR 3.3.3.2, Low Pressure ECCS Area Cooler Instrumentation, Rev. 0
Technical Requirements Manual and Basis TR 3.5.3, Equipment Area Coolers, Rev. 0
Work Order 113493095, 2-SR-3.5.1.6(CS I), Core Spray Flow Rate Loop I

Section 1EP6: Drill Evaluation

August 22, 2012, BFN August Training Drill Guide
BFN Performance Indicator Data, 2012 'A' Team Training Drill, August 22, 2012

Section 4OA1: Performance Indicator Verification

Browns Ferry Nuclear Plant, Unit 1 MSPI Basis Document, Revs. 10, 9, 8, 7, 6
Browns Ferry Nuclear Plant, Unit 2 MSPI Basis Document, Revs. 9, 8, 7, 6, 5
Browns Ferry Nuclear Plant, Unit 3 MSPI Basis Document, Revs. 8, 7, 6, 5
CDE #1044, EPIX #678, 1-FCV-73-45, HPCI System Testable Check Valve
CDE #1075, EPIX # 701, 1-PMP-73-29, HPCI Booster Pump
CDE #1161, 1-FCV-73-16, HPCI Steam Admission Valve
CDE #1192, 2-TRB-73-54, HPCI Excessive Unavailability
CDE #1211, EPIX #819, 1-FCV-73-18, HPCI Turbine Stop Valve
CDE #1250, 3-FCV-73-16, HPCI Unavailability
CDE #789, EPIX #527, 1-PCV-73-18C, HPCI Lube Oil Pressure Control Valve
MSPI Derivation Report, Browns Ferry Unit 1, for High Pressure Injection System, Unavailability Index (UAI), Periods; June 2011, Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 1, for High Pressure Injection System, Unreliability Index (URI), Periods; June 2011, Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 2, for High Pressure Injection System, Unavailability Index (UAI), Periods; June 2011, Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 2, for High Pressure Injection System, Unreliability Index (URI), Periods; June 2011, Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 3, for High Pressure Injection System, Unavailability Index (UAI), Periods; June 2011, Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 3, for High Pressure Injection System, Unreliability Index (URI), Periods; June 2011, Sept. 2011, Dec. 2011, Mar. 2012, June 2012
NPG-SPP-02.2, Performance Indicator Program, Rev. 4
PER 566687, Unplanned Inoperability and Reportability of U2 HPCI

Section 4OA2: Identification and Resolution of Problems

Annunciator Response Procedure for 1-XA-55-22C windows 5,6,7, and 8
Browns Ferry Various Thermography Images of Safety Related Operating Breakers
DCN 69995
ECI-0-000-BKR012, Checkout, Test and Installation of Masterpact NT Circuit Breakers, Rev. 5
Fort Calhoun Station Corrective Action Program Root Cause Analysis Report Breaker Cubicle 1B4A Fire, Rev. 2
LCOTR 0-052-OWA-2012-0169, Operator Workaround for seismic monitor modifications
Modified Mercalli Intensity Scale
NETA World Tech Brief, Plating of Contact Surfaces in Switchgear and Circuit Breakers, Winter 2005-2006
Nuclear Logistics Incorporated Response to the OPPD Switchgear Fire OE, Rev. 0, dated November 21, 2011
PER 523757, NRC Question of OE34373 – 480 Volt AC Switchgear Cubicle Fire
PER 546288, BFN Response to OE Report 34373 (Switchgear Fire at Fort Calhoun)
SR 525432, Inspection of Bus Stabs on 480v Shutdown Board 3B
WO 113351133, 3B 480V Board Inspection

WO 113359608, Service Building 480V Main Board A Rack and Inspect Five NT 800A and One NW 1600A Breaker
NPG-SPP-02.2, Performance Indicator Program, Rev. 4
0-OI-23, Residual Heat Removal Service Water System, Rev. 92
0-OI-67, Emergency Equipment Cooling Water System, Rev. 96
DWG 1-47E858-1, Rev. 64
DWG 2-47E858-1, Rev. 28
DWG 3-47E858-1, Rev. 33
DWG 1-47E859-1, Rev. 82
DWG 2-47E859-1, Rev. 31
DWG 3-47E859-1, Rev. 38
Browns Ferry Nuclear Plant, Unit 1 MSPI Basis Document, Revs. 10, 9, 8, 7
Browns Ferry Nuclear Plant, Unit 2 MSPI Basis Document, Revs. 9, 8, 7, 6
Browns Ferry Nuclear Plant, Unit 3 MSPI Basis Document, Revs. 8, 7, 6
MSPI Derivation Report, Browns Ferry Unit 1, for Cooling Water System, Unavailability Index (UAI), Periods; Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 1, for Cooling Water System, Unreliability Index (URI), Periods; Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 2, for Cooling Water System, Unavailability Index (UAI), Periods; Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 2, for Cooling Water System, Unreliability Index (URI), Periods; Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 3, for Cooling Water System, Unavailability Index (UAI), Periods; Sept. 2011, Dec. 2011, Mar. 2012, June 2012
MSPI Derivation Report, Browns Ferry Unit 3, for Cooling Water System, Unreliability Index (URI), Periods; Sept. 2011, Dec. 2011, Mar. 2012, June 2012
CDE #1059, 0-PMP-23-91, C3 EECW Pump
CDE #1063, EPIX # 687, 0-PMP-23-85, A3 EECW Pump
CDE #1064, EPIX # 692, 0-PMP-23-91, C3 EECW Pump
CDE #1065, 0-PMP-23-91, C3 EECW Pump
CDE #1066, 0-PMP-23-08, C1 RHRSW Pump
CDE #1099, 0-PMP-23-05, A2 RHRSW Pump
CDE #1177, B RHRSW Header
CDE #1183, 0-PMP-23-91, C3 EECW Pump
CDE #1207, EPIX # 811, 0-PMP-23-85, A3 EECW Pump
CDE #1214, 0-PMP-23-19, B2 RHRSW Pump

Section 4OA3: Event Follow-up

0-TI-230V, Vibration Program, Rev. 10
0-TI-471, Temporary Equipment Control, Rev. 06
2-47E812-1, Flow Diagram HPCI, Rev. 59
2-47E812-1-AppJ, Appendix J Testing Boundary for the HPCI system
3-AOI-100-1, Reactor Scram, Attachment 1, Scram Report, dated May 22, 2012
3-AOI-100-1, Reactor Scram, Attachment 1, Scram Report, dated May 29, 2012
BFN-50-7092, General Design Criteria Document, Neutron Monitoring System, Rev. 8
Browns Ferry – Emergency Diesel Generator System Vulnerability to Functional Failure Assessment, dated May 7, 2009
CDE 1041, Cause Determination Evaluation for MR ESF actuations

Dataware History for Unit 3 IRM signals from April 17 to April 26, 2013
Dataware History for Unit 3 IRM signals from April 8 to April 13, 2013
Design Criteria BFN-50-7082, Standby Diesel Generator, Rev. 16
Electro-Motive Vibration Guidelines Industrial Power Units, letter dated October 29, 1982
EMD Power Systems Owners Group Meeting, Diesel Generator Vibration Acceptable Criteria, dated June 26-28, 1991
EN 47787, Event Notification Report, March 29, 2012
Evaluation of the Significance of Combustibles Found in the BFN Unit 1, 2, and 3 Reactor Buildings 565 ft and 580 ft Elevations Red Zones, April 27, 2012
Fire Protection Report, Volume 1, Section 1, Fire Protection Plan, Rev. 12
Fire Protection Report, Volume 1, Section 2, Fire Hazards Analysis, Rev. 12
Fire Protection Report, Volume 1, Section 3, Safe Shutdown Analysis, Rev. 12
FSAR Section 8.5, Standby AC Power Supply and Distribution, BFN-24
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
LTAM BFN-11-0067, Separate IRM/SRM Pre-Amps into Separate J-Boxes
MREP meeting minutes from August 11, 2011
MREP meeting minutes from July 14, 2011
MREP meeting minutes from July 28, 2011
NPG-SPP-09.18.1, System Vulnerability Review Process (MCIP Reviews), Rev. 4
NPG-SPP-18.4.7, Control of Transient Combustibles, Rev. 02
NPG-SPP-6.9, Testing Programs, Rev. 0
NPG-SPP-6.9.1, Conduct of Testing, Rev. 5
NPG-SPP-6.9.3, Post-Modification Testing, Rev. 3
NRC Event Notification # 47955
NRC Memorandum to Douglas A. Broaddus: Audit Report Regarding Tennessee Valley Authority Browns Ferry Unit 1 AREVA Fuel Transition Emergency Core Cooling System Evaluation Model Application, dated August 16, 2011
OE25284 – Emergency Diesel Generator Governor Drive Oil Supply Line Sheared, North Anna 1 and 2
PER 135889, 3C IRM half scram
PER 144272, Unit 3 Reactor Scram while transferring power to 4kV Unit Board 3B
PER 164325, IRM 1D range switch causing half-scram
PER 213060, ADS Questions due to NRC Tech Spec Review
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
PER 349862, Valve checklist in error
PER 351673, Loss of Configuration Control / 2-LOV-073-0581
PER 362057, IRM 2G is spiking
PER 362395, Oil Leak Resulting in Emergency Shutdown of C DG
PER 373365, Full Scram due to SDV high water level Unit 3
PER 375372, Unit 2 ‘G’ IRM erratic
PER 381140, Maintenance Rule Plant Level Performance Criteria for unplanned ESF actuations
PER 402414 Downgrade request from B level apparent cause to C level document actions
PER 439393, IRM 3C has enough noise chatter that it is spiking and causing a half-scram
PER 465212, Green NRC-identified non-cited violation 05000259/2011004-01
PER 484548, Human Performance Shortfalls
PER 496575, Perform a fire loading analysis for in-use materials

PER 496580, Perform a fire loading analysis for in-use materials
PER 496581, Perform a fire loading analysis for in-use materials
PER 504005, Revised Analysis for Units 2 and 3 for Small Break LOCA
PER 505709, Potential Trend in the Human Performance Cross-Cutting Aspect H.2.c
PER 529001, Cables traverse the 20-foot red zones in the reactor buildings
PER 543131, 95003 Fundamental Problem: Technical Rigor
PER 556142, U3R15 Critical Path Delay
PER 556790, Design Error Resulting in Rework
PER 558437, Manual scram during Unit 3 reactor startup
PER 558488; Two of 13 Unit 3 MSRVs lift test outside acceptable range
PER 562103, BFN U3 Forced Loss Rate Indicator is Yellow for the Month of May 2012
PER 562343, Excessive Number of Unit 3 Unplanned Scrams
PER 563529, MMDP-14 [Rework] Evaluation for PER 558183
PER 564389, Trend of Vendor Design Technical Rigor and Oversight
PER 566687, Steam Packing Leak on BFN-2-FCV-73-81
PER 571054, U3 Scram on 5/22/12 is Determined to be OE Preventable
PER 571836, RCA 555573 Determined to be OE Preventable
PER 581990, Incorrect Design Process
PER 589759, Risk Rank of Contractor Prepared DCNs
PER 599571, DCN needed to install new bonnet on 20FCV-073-0081 HPCI warmup line
RCI-1.1, Radiation Operations Program Implementation, Rev. 149
RCI-25, Closed Circuit Television/Audio System, Rev. 07
SII-0-XX-92-054, IRM/SRM Testing and Temporary Protection Maintenance Instruction, Rev. 07
SR 565729, SR to document unplanned inoperability and reportability of U2 HPCI
SR 575293, RCA team recommendations for 2-FCV-73-81 WOs
SR 588295, Corrective Action Program Deficiencies
Technical Specifications and Bases Section 3.8, Electrical Power Systems, Amendment 280 and Rev. 52 respectively
TVA Letter to NRC: 10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Units 2 and 3, dated April 18, 2012
TVA Letter to NRC: 10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 2, dated August 20, 2012
PER 245385
PER 245386
PER 229734
PER 259787
PER 424389
PER 503304
Qualitative Risk Evaluation, PER 503304
BFN-0-12-032, Risk Evaluation For MSIV Failure

Section 4OA5: Other Activities
0-SR-DCS3.1.2.1, Spent Fuel Storage Inspection, Rev. 11
0-TI-508, Fuel Assembly Inspection Prior to MPC Loading, Rev. 3
0-TI-509, Spent Fuel Cask Loading Verification, Rev. 3
10 CFR 72.48 Screening Review, 0-SR-DCS3.1.2.1, Rev. 10 TN 0011
10 CFR 72.48 Screening Review, 0-SR-DCS3.1.2.1, Rev. 9 TN 0010

10 CFR 72.48 Screening Review, EDC 70586A, Use of HBF IAW Holtec CoC Amend. 5 and FSAR Rev. 7
10 CFR 72.48 Screening Review, EDC 70541 & Calculation NDQ007920050009, Rev. 2
10 CFR 72.48 Screening Review, 10CFR72.212 Rev. 4, List of Approved Cask Placed in Service
10 CFR 72.48 Screening Review, NFTP-100 Rev. 7
10 CFR 72.48 Screening Review, MSI-0-079-DCS200.1, Rev. 5
10 CFR 72.48 Screening Review, MSI-0-079-DCS200.2 Rev. 15
10 CFR 72.48 Screening Review, MSI-0-079-DCS200.2, Rev. 18
10 CFR 72.48 Screening Review, MSI-0-079-DCS200.2, Rev. 19
10 CFR 72.48 Screening Review, MSI-0-079-DCS200.2, Rev.21
10 CFR 72.48 Screening Review, MSI-0-079-DCS300.2, Rev. 1
10 CFR 72.48 Screening Review, MSI-0-079-DCS300.10, Rev 0
10 CFR 72.48 Screening Review, MSI-0-079-DCS300.10, Rev. 1
10 CFR 72.48 Screening Review, MSI-0-079-DCS300.11, Rev 0
10 CFR 72.48 Screening Review, MSI-0-079-DCS400.1, Rev. 2
10 CFR 72.48 Screening Review, MSI-0-079-DCS500.3, Rev. 3
10 CFR 72.48 Screening Review, MSI-0-079-DCS500.5, Rev. 3
10 CFR 72.48 Screening Review, WO 113165590
10 CFR 72.48 Screening Review, BFN 72.212 Evaluation Report Loading Campaign 6
10 CFR 72.212, Report of Evaluations, Rev. 5 dated 6/11/2012
Certificate of Compliance No. 1014, Appendix B, Design Features for the HI-STORM 100 Cask System, Section 3.6, Forced Helium Dehydration System, Amendment 5
CTP-SWD-100, Seismic Walkdowns EPRI 1025286, Rev. 0
Drawing 48NIO25 rev 3, Miscellaneous Steel Pipe Anchor Framing Sheet 1
EDC 70586, Allow Use of the FHD and SCS to Enable the Storage of High Burnup Fuel in the ISFSI, Rev. A
Flood Protection Walkdowns NEI 12-07, CTP-FWD-100, rev 0
HOLTEC HI STORM 100 Cask System, Safety Evaluation Report, Amendment 1
HOLTEC HI-STORM FSAR, Appendix 2.B, Forced Helium Dehydration System, Rev. 2
MSI-0-079-DCS035, Dry Cask Storage Campaign Guidelines, Rev. 11
MSI-0-079-DCS200.1, Dry Cask Preparations and Start Up, Rev. 5
MSI-0-079-DCS200.2, MPC Loading and Transport Operations, Rev. 25
MSI-0-079-DCS300.10, Forced Helium Dehydration System Operation, Rev. 2
MSI-0-079-DCS300.11, Supplemental Cooling System Operation, Rev. 0
MSI-0-079-DCS300.2, Alternate Cooling Water System Operation, Rev. 2
MSI-0-079-DCS400.1, ISFSI Abnormal Conditions Procedure, Placing the MPC in a Safe Condition, Rev. 2
MSI-0-079-DCS500.3, MPC Cooldown and Weld Removal, Rev. 3
MSI-0-079-DCS500.5, MPC Unloading Operations, Rev. 3
NFTP-100, Fuel Selection For Dry MPC Storage, Rev. 7
NFTP-113, FATF Development, Rev. 2
NFTP-113, FATF Development, Rev. 2
N-VT-4, System Pressure Test Visual Examination Procedure, Rev. 25
Procedure 0-AOI-100-3 revision 3, Flood Above 558’
SR 586141, Seismic Walkdowns - 1B SLC pump skid anchor bolt
SR584425, Design calculation CDQ0031894620
Temporary Instruction 2515/188 Seismic Walkdown and Area Walkby checklists

Transfer Operation No: BFN-1-119 MPC 0325
WO 113165590, PERFORM DRY CASK STORAGE OPERATION TO RELOCATE SPENT FUEL FROM THE UNITS 1 & 2 SPENT FUEL POOLS
Summary of Planned Transfer Operation No: BFN-1-119 MPC 0325
WO 113165582, Perform Dry Cask Storage Operation to Relocate Spent Fuel From the Units 1 & 2 Spent Fuel Pools
Vendor Procedure TRIVIS, Spent Fuel Cask Welding: HOLTEC Canisters GWS 9, Rev. 0

LIST OF ACRONYMS

ADAMS	-	Agencywide Document Access and Management System
ADS	-	Automatic Depressurization System
ARM	-	area radiation monitor
CAD	-	containment air dilution
CAP	-	corrective action program
CCW	-	condenser circulating water
CFR	-	Code of Federal Regulations
CoC	-	certificate of compliance
CRD	-	control rod drive
CS	-	core spray
DCN	-	design change notice
EECW	-	emergency equipment cooling water
EDG	-	emergency diesel generator
FE	-	functional evaluation
FPR	-	Fire Protection Report
FSAR	-	Final Safety Analysis Report
IMC	-	Inspection Manual Chapter
LER	-	licensee event report
MSRV		Main Steam Relief Valve
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