



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

June 9, 2014

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 – AMMENDED NRC INTEGRATED
INSPECTION REPORT 05000259/2012005, 05000260/2012005, AND
05000296/2012005**

Dear Mr. Shea:

This letter reissues Inspection Report 05000259/2012005, 05000260/2012005, and 05000296/2012005 (ADAMS Accession number ML13039A321) with an amendment to Section 4OA5.3. This amendment adds the independent inspection portion of Temporary Instruction (TI) 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns, that was performed by the NRC staff. This revision also identifies TI 2515/188 as Closed in the List of Items Opened, Closed and Discussed.

On December 31, 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 January 11, 2013, with Mr. Steve Bono, General Manager Site Operations, 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.

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

J. Shea

2

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

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: NRC Integrated Inspection Report 05000259/2012005,
05000260/2012005, and 05000296/2012005

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3

Letter to Joseph W. Shea from Jonathan H. Bartley dated June 9, 2014

SUBJECT: BROWNS FERRY NUCLEAR PLANT – AMMENDED NRC INTEGRATED
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05000296/2012005

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-259, 50-260, 50-296

License Nos.: DPR-33, DPR-52, DPR-68

Report No.: 05000259/2012005, 05000260/2012005, 05000296/2012005

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: October 1, 2012, through December 31, 2012

Inspectors: D. Dumbacher, Senior Resident Inspector
C. Stancil, Resident Inspector
P. Niebaum, Resident Inspector
L. Pressley, Resident Inspector
T. Stephen, Resident Inspector
D. Hardage, Resident Inspector
L. Suggs, Senior Construction Project Inspector (4OA2.5)
A. Sengupta, Reactor Inspector (1R08, 4OA5.3)
J. Laughlin, Emergency Preparedness Inspector (1EP4)
C. Kontz, Senior Project Engineer (4OA5.4)
G. Laska, Senior Operations Engineer (1R11.3)
K. Schaaf, Operations Engineer (1R11.3)
R. Baldwin, Senior Operations Engineer (1R11.4)

Approved by: Eugene F. Guthrie, Chief
Reactor Projects Special Branch
Division of Reactor Projects

Enclosure

SUMMARY

IR 05000259/2012005, 05000260/2012005, 05000296/2012005; 10/01/2012–12/31/2012; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Event Follow-up and Identification and Resolution of Problems

The report covered a three month period of inspection by the resident inspectors, three regional inspectors, and one headquarters inspector. Two licensee-identified violations of very low safety significance (Green) were identified. The significance of most findings is identified by their color (Green, White, Yellow, and Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 4, dated December 2006.

A. NRC Identified and Self-Revealing Findings

None

B. Licensee Identified Violations

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

Enclosure

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full Rated Thermal Power (RTP) except for one planned downpower to support the refueling outage (U1R9). On October 20, 2012 the unit was shutdown for a scheduled refueling outage that lasted 45 days. The unit was restarted on December 4, 2012 and returned to full power on December 7, 2012. The unit remained at near full power the remainder of the quarter.

Unit 2 operated at full RTP except for 3 planned downpowers and 1 unplanned SCRAM. On October 26, 2012, a planned downpower to 70 percent power was made for one day to complete Control Rod Sequence Exchange and SCRAM Time Testing. On November 23, 2012, a planned downpower to 95 percent power was made for Control Rod Sequence Exchange. On December 12, 2012, a planned downpower to 45 percent power was performed to enable maintenance on 2B recirculation pump Variable Frequency Drive, steam leak repairs to the 73-3 line, and repairs to 2B cond. booster pump. The plant returned to 100 percent power on December 14, 2012. On December 22, 2012, Unit 2 reactor automatically scrammed due to a post maintenance test failure associated with 3D emergency diesel and a wrong-train human performance error, respectively causing a loss of the 2B reactor protection subsystem and the 2A reactor protection subsystem. Unit 2 was restarted on December 25, 2012, and synchronized to the electrical grid on December 26, 2012. The unit remained at near full power the remainder of the quarter.

Unit 3 operated at full RTP power except for 3 planned downpowers. On October 15, 2012, a planned downpower to 60 percent power for one day to repair a steam leak on the 3A Feedwater drain line. On November 19, 2012, a planned downpower to 95 percent power for one day to repair a steam leak on 3C1/3C2 High Pressure Heaters. On December 14, 2012, a planned downpower to 60 percent power for control rod sequence exchange and turbine control valve testing. The unit remained at full power the remainder of the quarter.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection.1 Evaluate Readiness to Cope with External Floodinga. 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 Common intake structure Residual Heat Removal Service Water (RHRSW) pump rooms. 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 two External Flood Protection samples.

- Common Intake Structure RHRSW pump room hatches and vents as part of Temporary Instruction (TI) -187, Independent Flooding Walkdowns
- Licensee walkdown packages associated with RHRSW pump room and Diesel Generator building CO2 room watertight doors as part of Temporary Instruction (TI) - 187, Flooding Walkdowns

b. Findings

No findings were identified.

.2 Readiness for Seasonal Extreme Weather Conditionsa. Inspection Scope

Prior to and during the onset of cold weather conditions, the inspectors reviewed the licensee's implementation of 0-GOI-200-1, Freeze Protection Inspection, including applicable checklists: Attachment 1, Freeze Protection Annual Checklist; Attachment 2, Freeze Protection Operational Checklist; and as applicable, Attachments 3 through 12, Freeze Protection Daily Log Sheets for individual watch stations. The inspectors also reviewed the list of open FZ-coded Work Orders and Problem Evaluation Reports (PERs) to verify that the licensee was identifying and correcting potential problems relating to cold weather operations. In addition, the inspectors reviewed procedure requirements and walked down selected areas of the plant, which included the main control rooms, Residual Heat Removal Service Water (RHRSW) and Emergency Equipment Cooling Water (EECW) pump rooms, and all units Emergency Diesel Generator (EDG) buildings, to verify that affected systems and components were

Enclosure

properly configured and protected as specified by the procedure. The inspectors discussed cold weather conditions with Operations personnel to assess plant equipment conditions and personnel sensitivity to upcoming cold weather conditions. This constitutes one Readiness for Seasonal Extreme Weather sample.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

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

- Unit 1 Shutdown Cooling System
- Unit 1 Auxiliary Decay Heat Removal System
- Unit 3 Residual Heat Removal (RHR) System - Division II

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. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure NPG-SPP-18.4.6. Furthermore, the inspectors reviewed

Enclosure

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 Fire Protection inspection samples.

- Unit 1 Reactor Building, EL 593' 1B Electrical Board Room (Fire Area 4)
- Unit 2 Reactor Building, EL 593' 2B Electrical Board Room (Fire Area 8)
- Unit 3 Reactor Building, EL 519' through 639' (Fire Zone 3-1)
- Unit 3 Reactor Building, EL 621' 3A Electric Board Room (Fire Area 13)
- Unit 3 Reactor Building, EL 621' 3A 480 Shutdown Board Room, (Fire Area 14)

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

Non-Destructive Examination Activities and Welding Activities: From October 29, 2012, through November 1, 2012, the inspector conducted a review of the implementation of the licensee's In-service Inspection (ISI) Program for monitoring degradation of the reactor coolant system, emergency feedwater systems, risk-significant piping and components, and containment systems. The inspectors reviewed the implementation of the licensee's Risk Informed ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping boundaries. The inspectors activities consisted of an on-site review of NDE and welding activities to evaluate compliance with the applicable edition of the ASME Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2001 Edition through 2003 Addendum) and that indications and defects (if present) were appropriately evaluated, and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards or NRC approved alternative requirement.

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

- Directly observed:
 - Work Order # 112507860, Ultrasonic examination (UT) (manual) of Instrument Nozzle Safe End Welds in Feedwater System

- Reviewed records:
 - Work Order # 114009989, UT of HPCI System, pipe to flange
 - Work Order # 1-SI-4.6.G, Visual Examination of RHR System of Weld # 1-47B452H0158
 - Work Order # 1-SI-4.6.g, Magnetic particle Testing (MT) of RHR System of Weld # 1-47B452H0158-IA
 - Work Order # 114009989, Radiography Examination of HPCI system, Turbine Steam Supply Valve

During non-destructive surface and volumetric examinations performed since the previous refuelling outage, the licensee did not identify any recordable indications that required acceptance for continued service, therefore, no NRC review was required for this inspection procedure attribute.

The inspectors reviewed documentation for the repair/replacement of the following pressure boundary welds. The inspectors evaluated if the licensee applied the pre-service non-destructive examinations and acceptance criteria required by the construction Code. In addition, the inspectors reviewed the welding procedure specifications, welder qualifications, welding material certifications, and supporting weld procedure qualification records to evaluate if the weld procedures were qualified in accordance with the requirements of the Construction Code and the ASME Code Section IX.

- Welding package for HPCI 2” Turbine Exhaust at Drain (Work Order # 112453386)
- Welding package for HPCI System, pipe to flange (Work Order # 114009989)

Identification and Resolution of Problems: The inspectors performed a review of ISI-related problems, including welding that were identified by the licensee and entered into the Corrective Action Program (CAP) as Condition Report (CRs). The inspectors reviewed the CRs to confirm that the licensee had appropriately described the scope of the problem, description of the evaluation and had identified appropriate corrective actions. The review also included the review of the licensee’s use, consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. Document reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On October 17, 2012, the inspectors observed a licensed operator requalification simulator examination for an operating crew according to a Unit 2 Simulator Exercise Guide, (SEG), scenario which contained at a minimum the following attributes; main generator hydrogen leak, loss of offsite power (LOOP), inadvertent pump start, failure to scram, unisolable reactor core isolation cooling (RCIC) system leak, fuel element failure.

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

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of procedures including Abnormal Operating Instructions (AOIs), and Emergency Operating Instructions (EOIs)
- Timely control board operation and manipulation, including high-risk operator actions
- Timely oversight and direction provided by the shift supervisor, including ability to identify and implement appropriate technical specifications actions such as reporting and emergency plan actions and notifications
- Group dynamics involved in crew performance

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

.3 Biennial Licensed Operator Requalification

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of October, 8-11, 2012, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the facility licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1985, "American National Standard for Nuclear Power Plant Simulators for use in Operator Training and Examination." The inspectors observed two shift crews during the

Enclosure

performance of the operating tests. Documentation reviewed included written examinations, Job Performance Measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records, simulator performance test records, operator feedback records, licensed operator qualification records, remediation plans, watchstanding records, and medical records. The records were inspected using the criteria listed in Inspection Procedure 71111.11. Documents reviewed are listed in the Attachment.

The inspectors selected PER 245312 "Reactivity Management [Control] Plan (RCP) requires improvement" for a detailed review. PER 245312 states: Review of a completed RCP for the Unit 3 down power and shutdown and other RCPs indicated a lack of rigor in the execution of some of the RCP steps. Incomplete guidance on a number of the RCP steps, and some knowledge discrepancies exist. The Quality Assurance group recommended that additional training be developed for operators and to improve implementing reactivity management plans. The training department developed training on reactivity management plans that was delivered to the operations group. The reactor engineering group was invited to attend these training sessions to add additional technical knowledge. Inspectors reviewed the training presentations developed for the additional training. It appears this training was effective.

b. Findings

No findings were identified.

.4 Annual Review of Licensee Requalification Examination Results

a. Inspection Scope

On December 12, 2012, the licensee completed the annual requalification operating examinations required to be administered to all licensed operators in accordance with 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the individual operating examinations and the crew simulator operating examinations in accordance with Inspection Procedure (IP) 71111.11, "Licensed Operator Requalification Program." These results were compared to the thresholds established in Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Appendix I, "Operator Requalification Human Performance Significance Determination Process."

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness.1 Routinea. Inspection Scope

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

- Browns Ferry Valve Stem Packing Program
- Control Room Emergency Ventilation System

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluationa. Inspection Scope

For planned online work and/or emergent work that affected the combinations of risk significant systems listed below, the inspectors examined four on-line maintenance risk assessments, and actions taken to plan and/or control work activities to effectively manage and minimize risk. The inspectors verified that risk assessments and applicable risk management actions (RMA) were conducted as required by 10 CFR 50.65(a)(4) applicable plant procedures, and BFN Equipment to Plant Risk Matrix. Furthermore, as applicable, the inspectors verified the actual in-plant configurations to ensure accuracy of the licensee's risk assessments and adequacy of RMA implementations. This activity constituted four Maintenance Risk Assessment inspection samples.

- October 19, 2012, Unit 3 RPS 3A Motor Generator Set Failure with 3E Raw Cooling Water Pump and G Control and F Service Air Compressors Out of Service (OOS)
- October 25, 2012, Units 1 and 2, C Shutdown Board, C EDG and RHRSW Pumps B2 and B3 OOS, 2B CCW Pump OOS, Unit 1 ORAM Yellow
- November 1, 2012, Unit 1 Orange planned outage risk associated with local leak rate test of valve 74-68 (Operation with Potential to Drain the Reactor Vessel, OPDRV)
- December 13, 2012, Unit 2 elevated Green risk for Single Loop Operations

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed Technical Specification operability. The inspectors also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedure NEDP-22, Functional Evaluations, and NEDP-27, Past Operability Evaluations, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors examined the implementation of compensatory measures to verify that they achieved the intended purpose and that the measures were adequately controlled. The inspectors also reviewed PERs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. This activity constituted six Operability Evaluation inspection samples.

- Emergency Equipment Cooling Water South Header Combined Leakage of D Strainer and B3 Pump (PERs 617833 and 617840)
- 3C Emergency Diesel Generator Fast Start and Load Sharing Relay Configurations (PER 617890)
- Reactor Building South Access Watertight Door Broken Frame Weld (PER 623264)
- High Pressure Coolant Injection (HPCI) Steam Line Inboard Isolation Valve Failure Due to Inadequate Manufacturer's Assembly (PER 639155)
- Residual Heat Removal Service Water (RHRSW) Pump Room Sump Debris (PER 618735)
- Unit Common Standby Gas Treatment Train C Inoperable longer than allowed by Technical Specification (PERs 590208 and 604350)

b. Findings

No findings were identified.

1R18 Plant Modifications.1 Permanent Plant Modificationsa. Inspection Scope

The inspectors reviewed the Design Change Notice (DCN) and completed work package (WO 113086731) for DCN 70488, Replace Unit 1 HPCI 73-16 gate valve with a different design gate valve, including related documents and procedures. The inspectors reviewed licensee procedures NPG-SPP-9.3, Plant Modifications and Engineering Change Control, and NPG-SPP-6.9.3, Post-Modification Testing, and observed part of the licensee's activities to implement this design change made while the unit was online. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify that the modifications had not affected system operability/availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents.

b. Findings

No findings were identified.

1R19 Post Maintenance Testinga. Inspection Scope

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

- Unit 1: Residual heat removal service water (RHRSW) Pump B1 Hand switch, (0-HS-023-0015A/3), replacement per WO 111436112
- Unit 1: High Pressure Coolant Injection (HPCI) Turbine Steam Supply Valve, (1-FCV-073-0016), per WO 113657859, MOVATS Testing per ECI-0-000-MOV9; and WO 113195490, HPCI Comprehensive surveillance per 1-SR-3.5.1.7(COMP)

- Unit 3: 3C Emergency Diesel Generator Fast Start and Load Sharing Relay Configurations (PER 617890) per WO 114009014 and Procedure 3-SR-3.8.1.1(3C), Diesel Generator 3C Monthly Operability Test
- Unit 3: 3A Residual Heat Removal (RHR) pump breaker and cubicle preventive maintenance per WO 113690981 and 3-SR-3.5.1.6 (RHR-I)

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 1 Scheduled Refueling Outage (U1R9)

a. Inspection Scope

From October 20, 2012, through December 4, 2012, the inspectors examined critical outage activities to verify that they were conducted in accordance with Technical Specifications, applicable plant procedures, and the licensee's outage risk assessment and management plans. The inspectors also monitored critical plant parameters, and observed operator control of plant conditions, during Cold Shutdown (Mode 4), Startup (Mode 2), and Power Operation (Mode 1). Some of the significant outage activities specifically reviewed and/or witnessed by the inspectors were as follows:

Outage Risk Assessment

Prior to the U1R9 refueling outage that began on October 20, the inspectors attended outage risk assessment team meetings and reviewed the Outage Risk Assessment Report to verify that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing an outage plan that assured defense-in-depth of safety functions were maintained. The inspectors also reviewed the daily U1R9 Refueling Outage Reports, including the Outage Risk Assessment Management (ORAM) Safety Function Status, and regularly attended the twice a day outage status meetings. These reviews were compared to the requirements in licensee procedure SPP-7.2, Outage Management, and TS. These reviews were also done to verify that for identified high risk significant conditions, due to equipment availability and/or system configurations, contingency measures were identified and incorporated into the overall outage and contingency response plan. Furthermore, the inspectors frequently discussed risk conditions and designated protected equipment with Operations and outage management personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process

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

Decay Heat Removal

The inspectors reviewed licensee procedures 1-OI-74, Residual Heat Removal System (RHR); 1-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Alternate Decay Heat Removal System Failures; and conducted a main control room panel and in-plant walkdowns of system and components to verify correct system alignment. During planned evolutions that resulted in increased outage risk conditions for shutdown cooling, inspectors verified that the plant conditions and systems identified in the risk mitigation strategy were available. In addition, the inspectors reviewed controls implemented to ensure that outage work was not impacting the ability of operators to operate spent fuel pool cooling, RHR shutdown cooling, and/or Alternate Decay Heat Removal (ADHR) system. Furthermore, the inspectors conducted several walkdowns of the ADHR system during operation with the fuel pool gates removed.

Critical Outage Activities

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

- Walked down selected safety-related equipment clearance orders (i.e., tag order 1-TO-2012-0003, sections 1-074-0016 and 1-074-0017A for 1D RHR Pump motor and rotating element replacements)
- Verified Reactor Coolant System (RCS) inventory controls, specifically the November 1st evolution supporting RHR valve local leak rate testing which had the potential to drain the reactor vessel (OPDRV), were controlled per 1-POI-200.5
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) and TS compliance during the various shutdown modes of operation, and mode transitions
- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations and overall integrity
- Examined foreign material exclusion controls particularly in proximity to and around the reactor cavity, equipment pit, and spent fuel pool
- Routine tours of the control room, reactor building, refueling floor and drywell
- Verified the licensee was managing fatigue by review of fatigue assessments and review of certain outage and non-outage workers' schedules and work hours (There were no waiver requests or self declarations.)

Enclosure

Reactor Vessel Disassembly and Refueling Activities

The inspectors witnessed selected activities associated with reactor vessel disassembly, and reactor cavity flood-up and drain down in accordance with 1-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, on numerous occasions, the inspectors witnessed fuel handling operations during the two Unit 1 reactor core fuel shuffles performed in accordance with TS and applicable operating procedures, such as 0-GOI-100-3A, Refueling Operations (In Vessel), 0-GOI-100-3B, Operations in the Spent Fuel Pool, and 0-GOI-100-3C, Fuel Movement Operations During Refueling. The inspectors verified specific fuel movements as delineated by the Fuel Assembly Transfer Sheets (FATF). Furthermore, the inspectors also witnessed and examined the video verification of the final completed reactor core conducted per Attachment 6, of 0-GOI-100-3C.

Torus Closeout

On November 24, 2012, the inspectors reviewed the licensee's conduct of 1-GOI-200-2, Torus Closeout, and performed an independent detailed closeout inspection of the Unit 1 Torus.

Drywell Closeout

On November 27, 2012, the inspectors reviewed the licensee's conduct of 1-GOI-200-2, Drywell Closeout, and performed an independent detailed closeout inspection of the Unit 1 drywell.

Restart Activities

The inspectors specifically conducted the following:

- Witnessed heatup and pressurization of Unit 1 reactor pressure vessel in accordance with 1-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor pressure Vessel and Associated Piping.
- Evaluated licensee actions and response to reactor vessel inner o-ring leakage.
- Reviewed Primary Containment Total Leak Rate results
- Witnessed Unit 1 approach to criticality and power ascension per 1-GOI-100-1A, Unit Startup, and 1-GOI-100-12, Power Maneuvering
- Reactor Coolant Heatup/Pressurization to Rated Temperature and Pressure per 1-SR-3.4.9.1, Reactor Heatup and Cooldown Rate Monitoring
- Evaluated licensee decision (December 2, 2012, ODMI/PER 651334) to perform plant startup and operate plant with an existing reactor vessel inner o-ring leak

Corrective Action Program

The inspectors reviewed PERs generated during U1R9 and attended management review committee (MRC) meetings to verify that initiation thresholds, priorities, mode holds, operability concerns and significance levels were adequately addressed. Resolution and implementation of corrective actions of several PERs were also reviewed for completeness.

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions of, and/or reviewed completed test data for the following surveillance tests of risk-significant and/or safety-related systems to verify that the tests met TS surveillance requirements, UFSAR commitments, and in-service testing and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the SSCs were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement. This activity constituted five Surveillance Testing inspection samples, one in-service, two routine tests, one containment isolation test, and one reactor coolant system leak detection test.

In-Service Tests:

- November 19, 2012, 1-SR-3.1.7.7, Standby Liquid Control System Functional Test

Routine Surveillance Tests:

- October 23, 2012, 0-SR-3.8.1.9C, C Emergency Diesel Generator Load Acceptance Test
- November 28, 2012, 1-SR-3.5.1.8, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at 150 psig Reactor Pressure

Containment Isolation Valve Tests:

- November 1, 2012, Local Leak Rate Test of Unit 1 Residual Heat Removal valves 74-67 and 74-68

Reactor Coolant System Leak Detection Tests:

- December 6, 2012, 3-SI-4.2.E-1(B), Drywell Equipment Drain Sump Flow Integrator Calibration

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The NSIR headquarters staff performed an in-office review of the latest revisions of various Emergency Plan Implementing Procedures (EPIPs) and the Emergency Plan located under ADAMS accession numbers ML12296A649, ML12307A285, and ML12199A022 as listed in the Attachment.

The licensee determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection. Documents reviewed are listed in the Attachment. This inspection activity satisfied one inspection sample for the emergency action level and emergency plan changes on an annual basis.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Initiating Events

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the PIs listed below, including procedure NPG-SPP-02.2. The inspectors examined the licensee's PI data for the specific PIs listed below for the fourth quarter of 2011 through the third quarter of 2012. The inspectors compared the licensee's raw data against graphical representations and specific values reported to the NRC for the third quarter of 2012 to verify that the data was correctly reflected in the report. Furthermore, the inspectors validated this data against relevant licensee records (e.g., PERs, Daily Operator Logs, Plan of the Day, LERs, 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, to ensure that industry reporting guidelines were appropriately applied. This activity constituted nine Performance Indicator Verification inspection samples; three unplanned scrams, three Unplanned Scrams with Complications, and three Unplanned Power Changes.

Enclosure

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

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

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

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing daily PER and Service Request (SR) reports, and periodically attending Corrective Action Review Board (CARB) and PER Screening Committee (PSC) meetings.

.2 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors reviewed the specific corrective actions associated with failed GE HFA Relays on Unit 2 (PER 618132) and Unit 3 (PER 571080).

b. Assessment and Observations

The inspectors had the following observations:

The inspectors reviewed the impact, causal analysis, and corrective actions for five AC HFA relay failures that resulted in "half-scrams" from 2003 to 2012. The licensee's apparent cause analysis for the failures concluded the relays were an older style and had thus failed due to insulation breakdowns. After the inspectors questioned this conclusion, the licensee determined that the glass enclosure for the relays which retained heat generated had not been factored into their life expectancy. Additionally, the inspectors questioned the licensee's application of previous guidance related to replacement of HFA relays. Licensee follow-up identified that additional inspections were needed to ensure installed HFA relays were of a type capable of operating in the enclosure device.

Browns Ferry conducted walkdowns of approximately 2275 HFA relays on the three units to determine whether relays required replacement. Forty Four relays had to be replaced.

c. Findings

No findings were identified.

.3 Semiannual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP implementation and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review included the results from daily screening of individual PERs (see Section 40A2.1 above), licensee trend reports and trending efforts, and independent searches of the PER database and WO history. The inspectors' review nominally considered the six-month period of July 2012 through December 2012, although some searches expanded beyond these dates. Additionally, the inspectors' review also included the Integrated Trend Reports (ITR) from the third and fourth quarters of fiscal year 2012. The licensee reports covered the period of April 1, 2012 to September 30, 2012. Furthermore, the inspectors verified that adverse or negative trends identified in the licensee's PERs, periodic reports and trending efforts were entered into the CAP. Inspectors interviewed the appropriate licensee staff and also reviewed procedures, NPG-SPP-02.8, Integrated Trend Review and NPG-SPP-02.7, PER Trending.

The purpose of the licensee's integrated trend reviews was to identify the top site and departmental issues (gaps to excellence) requiring management attention. Other objectives were to provide status of the top issues and their progress to resolution, identify continuing issues, emerging trends and issues to be monitored, review progress towards resolving past top issues, review issues identified by external organizations such as the NRC, INPO, Nuclear Safety Review Board (NSRB), QA, etc., and determine why they were not identified by line organizations.

b. Findings and Observations

No findings were identified. However, the inspectors had the following observations discussed below:

Inspectors noted that licensee-identified third and fourth quarter Corrective Action Program (CAP) and Human Performance issues continued from the first and second quarter 2012 trend reports. These issues were also identified as Fleet and Site top priorities. The majority of the key actions to resolve the gaps for these issues were still in progress. Trending information provided in the fourth quarter fiscal year 2012 report showed evidence of improvement in the CAP and Human Performance metrics. The site's Human Performance error rate metrics were indicating the best of the last two years.

Enclosure

In addition to reviewing the site's progress on the above issues, the inspectors conducted an independent review of the licensee's CAP to identify potential adverse trends.

4. Focused Annual Sample Review - Operator Workarounds

a. Inspection Scope

The inspectors conducted a review of existing Operator Workarounds (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. The inspectors examined all active OWAs listed in the Limiting Condition of Operation Tracking (LCOTR) Log, and reviewed them against the guidance in BFN-ODM-4.16, Operator Workarounds/Burdens/Challenges. The inspectors also discussed these OWAs 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 selected OWAs, and verified the ongoing performance, and/or feasibility of, the required actions. Lastly, for selected OWAs, the inspector reviewed the applicable PER, including the associated functional evaluation and corrective action plans (both interim and long term).

b. Findings and Observations

No findings were identified. However, the inspectors had the following observations which were discussed with the licensee:

Inspectors determined that, in general, Browns Ferry adequately tracks and trends all operator workarounds, burdens and challenges. This includes estimating, tracking and compiling the aggregate impact of the workarounds, burdens and challenges. Inspectors identified multiple occasions where operations staff had routinely updated the progress of corrective actions within the confines of the software program designed to track OWA's (eSOMS). The licensee did self-identify that workarounds from equipment failures have adversely affected the time available for operators to perform their normal duties.

Inspectors reviewed trending information on workarounds, burdens and challenges which was reported weekly within the Browns Ferry Plan of the Day. Unit 1 and unit common systems both had 2 OWA's and were well above the goal for the majority of the year. Unit 3, however, spent the majority of the year with a high number of operator burdens. Common unit burdens also finished the year well above the station's goals.

Inspectors noted that workarounds, burdens, and challenges were existing beyond one operating cycle on a unit. There were examples of multiple burdens and challenges that were not driven to conclusion prior to plant restart and were allowed to remain outstanding following refueling outages. Inspectors determined that the site often fails to adequately address and drive to resolution lower level operator issues when the opportunity is available to do so.

Enclosure

The licensee entered all the above issues into the CAP as SR 665557.

.5 Focused Annual Sample Review – Environmental Qualification (EQ)

a. Inspection Scope

Inspectors conducted a review of the licensee's EQ program to verify that the licensee was identifying and resolving problems associated with EQ equipment at an appropriate threshold, entering them into the corrective action program, establishing adequate compensatory measures, prioritizing resolution of the problems, and implementing appropriate corrective actions in a timely manner commensurate with its safety significance. During the week of December 17, the inspectors interviewed engineering personnel and reviewed a sample of the licensee's electronic EQ binders, EQ program assessments, health reports, environmental qualification information releases (EQIRs) and EQ related problem evaluation reports (PERs) to ensure general EQ program and 10 CFR 50.49 adherence. Inspectors also conducted walkdowns of a sample of accessible EQ equipment to ensure configuration control was being maintained and that equipment was installed in accordance with the tested configuration.

b. Findings

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

4OA3 Event Follow-up

.1 (Closed) Licensee Event Reports (LERs) 05000259, 260, and 296/2012-008-00 and -01, Standby Gas Treatment System Train C Inoperable Longer Than Allowed by Technical Specifications

a. Inspection Scope

On August 2, 2012, the licensee placed the common Standby Gas Treatment (SGT) Train C relative humidity heater into service which resulted in an annunciator alarm indicating power was lost to the SGT Train C filter bank heater element. Licensee troubleshooting identified that the motor control center (MCC) bucket containing the associated breaker was misaligned due to a missing retaining device. The licensee concluded that the SGT Train C had been inoperable since preventive maintenance performed in September, 2011, as the result of an inadequate maintenance instruction which allowed installation of a breaker bucket with a single retaining device. The inspectors reviewed the initial LER issued on October 1, 2012, the LER revision issued on December 14, 2012, and associated Problem Evaluation Report (PER) 604350, which included the cause determination and corrective action plans. These licensee evaluations concluded that the relative humidity heater was not required for the SGT Train C to perform accident required functions.

b. Findings

No findings were identified. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 05000296/2012-004-00, Manual Reactor Scram During Startup Due to Multiple Control Rod Insertion

a. Inspection Scope

The inspectors reviewed the LER for potential performance deficiencies and/or violations of regulatory requirements. The LER was associated with the Unit 3 manual reactor scram that occurred during a reactor startup on May 24, 2012. The inspectors reviewed the root cause report associated with this event and discussed the issue with appropriate members of plant staff. The cause of the scram was attributed to Unit Operator error combined with IRM signal spikes associated with manipulation of the scram reset switch and a degraded IRM High Voltage coaxial cable connector on the 3A IRM. This condition was documented in the licensee's corrective action program as PER 558437. Additional documents reviewed are listed in the Attachment. This LER is closed.

b. Findings

No findings were identified. This LER is closed

.3 (Closed) Licensee Event Reports (LER) 05000296/2012-006-00; 05000296/2012-006-01, Main Steam Relief Valves Lift Settings Outside Technical Specification Required Setpoint

a. Inspection Scope

The inspectors reviewed LER 05000296/2012-006-00 and 05000296/2012-006-01 dated July 24, 2012, and August 31, 2012, and the applicable PER 558488. On May 25, 2012, two of thirteen Browns Ferry Nuclear Plant Unit 3 main steam relief valves, during testing, had mechanically actuated at pressures outside the allowed +/- percent tolerance per Technical Specification 3.4.3 setpoint. One relief valve lifted high at + 3.98 percent and the other low at negative 3.1 percent. This Technical Specification Limiting Condition for Operation required 12 of the S/RVs to be capable to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function). The licensee's analysis concluded that the variations in lift setting pressures did not prohibit the ability of the MSRVs to perform the function to open in order to provide over pressure protection. Twelve S/RVs were available to relieve excess pressure if the setpoint had been exceeded. However, contrary to the technical specifications surveillance requirement, only 11 operable main steam relief valves passed the licensee lift test procedure. The root cause was determined by the Tennessee Valley Authority to be that the valve design does not make allowances for corrosion bonding. Browns Ferry captured the corrective actions in PER 558488.

b. Findings

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

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings were identified.

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

a. Inspection Scope

Inspectors conducted independent walkdowns to verify that the licensee completed the actions associated with the flood protection feature specified in paragraph 03.02.a.2 of this TI. Inspectors are performing walkdowns 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 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, and are available, functional, and properly maintained.

Enclosure

b. Findings

Findings or violations associated with the flooding, if any, will be documented in the 1st quarter integrated inspection report of 2013.

.3 (Closed) Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied the licensee on their seismic walkdowns of Unit 1 HPCI, Unit 2 'D' Shutdown Board Room and Unit 3 Core Spray Loop I on August 8, 2012, and verified that the licensee confirmed that the following seismic features associated with Unit 1 HPCI, Unit 2 'D' 4kV shutdown board, 2B 250VDC RMOV board, 2B 480V RMOV board and Unit 3 'A' and 'C' Core Spray Pumps were free of potential adverse seismic conditions.

The inspectors also verified that the licensee confirmed that the following seismic features associated with the above components were free of potential adverse seismic conditions:

- Anchorage was free of bent, broken, missing or loose hardware
- Anchorage was free of corrosion that is more than mild surface oxidation
- Anchorage was free of visible cracks in the concrete near the anchors
- Anchorage configuration was consistent with plant documentation.
- SSCs will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

On October 17 – 19, 2012, the inspectors independently performed their walkdown of Unit 3 Diesel Generator A, B, Units 1, 2 Residual Heat Exchanger Pumps, Units 1, 2 Residual Heat Exchangers, Unit 3 Control Room CR Water Chiller, Unit 1 Service Water Pump.

The inspectors also verified that the licensee confirmed that the following seismic features associated with the above components were free of potential adverse seismic conditions:

- Anchorage was free of bent, broken, missing or loose hardware
- Anchorage was free of corrosion that is more than mild surface oxidation
- Anchorage was free of visible cracks in the concrete near the anchors
- Anchorage configuration was consistent with plant documentation.
- SSCs will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

The inspectors also looked at the items from the Area Walk By (AWB) list:

- Unit 3 Diesel Generator A, B
- Unit 3 Control Room Water Chiller
- Unit 1 Service Water Pump

Observations made during the walkdown that could not be determined to be acceptable were entered into the licensee's corrective action program for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the SWEL and these items were walked down by the licensee. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

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

a. Inspection Scope

During the inspection period the inspectors performed a follow-up review of TVA's implementation of Confirmatory Order for Office of Investigation Report Nos. 2-2006-025 & 2-2009-003, item number 1. This item is closed.

1. By no later than ninety (90) calendar days after the issuance of this Confirmatory Order, TVA shall implement a process to review proposed licensee adverse employment actions at TVA's nuclear plant sites before actions are taken to

Enclosure

determine whether the proposed action comports with employee protection regulations, and whether the proposed actions could negatively impact the SCWE.

During the inspection period the inspectors performed a follow-up review of TVA's implementation of Confirmatory Order for Office of Investigation Report Nos. 2-2006-025 & 2-2009-003, item numbers 4, 6, and 10. These items are not closed.

4. Through the end of calendar year 2013 and on approximately a quarterly basis, TVA shall continue to analyze SCWE trends and develop planned actions, as appropriate
6. Through calendar year 2013, TVA shall conduct "Town Hall"-type meetings at least annually at its nuclear power plants and corporate office with TVA and contractor employees which address topics of interest, including a discussion on TVA's policy regarding fostering a SCWE.
10. TVA's annual online computer-based training course initiative, which discusses the components of a nuclear safety culture, what is meant by a SCWE, and the avenues available to raise concerns, shall be maintained through calendar year 2013.

b. Findings and Observations

No findings were identified.

40A6 Meetings, Including Exit

.1 Exit Meeting Summary

On January 11, 2013, the resident inspectors presented the quarterly inspection results to Mr. S. Bono, General Plant Manager, Site Operations, 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.

40A7 Licensee-Identified Violations

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

- The licensee-identified a violation of 10 CFR Part 50, Appendix B, Criterion XVI for the licensee's failure to assure that conditions adverse to quality, such as deficiencies, and nonconformances are promptly identified and corrected. Specifically, the licensee failed to take timely corrective actions to address an extensive backlog of EQ information releases which resulted in not meeting their environmental qualification program and the 10CFR 50.49 auditability requirements. Contrary to this requirement, since January of 2010, the licensee failed to take prompt and appropriate corrective actions to evaluate and correct an extensive backlog of EQIRs, which resulted in 81 of the licensee's 99 required Environmental Qualification equipment files not being updated to reflect the as-installed specifications and configuration of EQ equipment. The licensee entered this issue

Enclosure

into their corrective action program as PERs 238931 and 624137. The finding was determined to be of very low safety significance (Green) using Attachment 4 to IMC 0609, Significance Determination Process, because the incomplete corrective actions did not result in an actual loss of safety function.

- Unit 3 Technical Specification 3.4.3, Safety/Relief Valves, required that twelve of thirteen main steam safety relief valves (MSRVs) lift at a setpoint within plus or minus three percent of a specified value. Contrary to this, during TS required surveillance testing following the Unit 3 Cycle 9 refueling outage, the licensee discovered that the lift setpoints of two MSRVs exceeded the plus or minus three percent TS allowed pressure band. This TS violation was entered into the licensee's CAP as PER 558488. The finding was determined to be of very low safety significance because the as-found lift setpoint conditions of the Unit 3 MSRVs were evaluated and determined to meet the design basis criteria for the most limiting reactor pressure vessel over-pressurization events.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Bono, General Manager Site Ops
J. Boyer, Assistant Director of Engineering
E. Cobey, Licensing
J. Davenport, Licensing
G. Dudley, Site Welding/Repair & Replacement
M. Ellet, Maintenance Rule Coordinator
J. Emens, Nuclear Site Licensing Manager
F. Froscello, ISI Program
W. Hayes, Reactor Engineering Manager
M. Henderson, Vessel Internals Program
H. Higgins, LOR Supervisor (Acting)
L. Hughes, Operations Manger
M. Hunter, Mechanical Maintenance Manager
D. Kettering, Electrical Systems Engineering Manager
T. McCaney, Operations
B. McCreary, Senior Program Manager, Employee Concerns
J. McCormack, Ventilation Systems Engineer
F. Nielson, IWE/IWL Programs
M. Oliver, Site Licensing
K. Polson, Site Vice President
M. Rasmussen, W.C. Manager
T. Scott, PI Manager
R. Stowe, Equipment Reliability Manager
J. Shea, Vice President Nuclear Licensing
P. Summers, DSL
C. Vaughn, Operations Training Manager
M. Webb, Site Licensing
M. Wilson, Site Training Direct

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Closed

05000259, 260, and 296/2012-008-00	LER	Standby Gas Treatment System Train C Inoperable Longer Than Allowed by Technical Specifications (Section 4OA3.1)
05000259, 260, and 296/2012-008-01	LER	Standby Gas Treatment System Train C Inoperable Longer Than Allowed by Technical Specifications (Section 4OA3.1)
05000296/2012-004-00	LER	Manual Reactor Scram During Startup Due to Multiple Control Rod Insertion (Section 4OA3.2)
05000296/2012-006-00	LER	Main Steam Relief Valves Lift Settings Outside Technical Specification Required Setpoint (Section 4OA3.3)
05000296/2012-006-01	LER	Main Steam Relief Valves Lift Settings Outside Technical Specification Required Setpoint (Section 4OA3.3)
05000259, 260, 296- 00	ORD	12/29/2009 Confirmatory Order Action 1 (Section 4OA5.4)
2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns (Section 4OA5.3)

Discussed

2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns (Section 4OA5.2)
05000259, 260, 296- 00	ORD	12/29/2009 Confirmatory Order Action 4 (Section 4OA5.4)
05000259, 260, 296- 00	ORD	12/29/2009 Confirmatory Order Action 6 (Section 4OA5.4)
05000259, 260, 296- 00	ORD	12/29/2009 Confirmatory Order Action 10 (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

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

NPG-SPP-10.14, Freeze Protection, Rev. 0
47W225-16, Diesel Generator Building Units 1-3, Environmental Data EI 583.5, Rev. 4
47W225-17, Diesel Generator Building Units 1&2, Environmental Data EI 565.5, Rev. 4
47W225-18, Diesel Generator Building Unit 3, Environmental Data EI 565.5, Rev. 4
47W225-19, Diesel Generator Building Unit 3, Environmental Data EI 583.5, Rev. 4
PCR 12003545, Remove Abandoned Equipment from Freeze Protection GOI
PER 661731, Freeze Protection GOI Refers to Breaker With No Landed Field Wiring
PER 661742, Freeze Protection GOI Attachment 1 Documented Removed Piping
PER 661745, Freeze Protection GOI Attachment 1 References Abandoned Equipment
PER 661747, Operator Incorrectly Initialed D EDG room space heater
CTP-FWD-100, Flood Protection Walkdowns NEI 12-07, Rev. 0
NEI 12-07, Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features, Rev. 0-A
WO 113618794, Perform Flood Protection Walkdowns IAW CTP-FWD-100
DWG 31N203, Concrete Pumping Station Outline – Sheet 1, Rev. 8
0-AOI-100-3, Flood Above Elevation 558', Rev. 35
PER 637130, Flood Walkdowns – Preventive Maintenance Hatches and Manways

Section 1R04: Equipment Alignment

PIP 95-71 Reactor Level and Pressure Instrumentation
1-47E811-1 Flow Diagram Residual Heat Removal System
0-OI-72 Auxiliary Decay Heat Removal (ADHR) System Operations
0-OI-72/ATT-1 ADHR System Valve Lineup Checklist
0-OI-72/ATT-2 ADHR System Panel Lineup Checklist
0-OI-72/ATT-3 ADHR System Electrical Lineup Checklist
0-OI-72/ATT-4 ADHR Instrument Inspection Checklist
0-15E900-1 Electrical Instrument Details
0-47E610-72-1 Control Diagram ADHR System Sheet 1
0-47E610-72-2 Control Diagram ADHR System Sheet 2
0-47E873-1 Flow Diagram ADHR Sheet 1
0-47E873-2 Flow Diagram ADHR Sheet 2
3-47E811-1, Flow Diagram Residual Heat Removal System, Rev. 67
3-OI-74/ATT-1, Valve Lineup Checklist Unit 3, Rev. 87
3-OI-74/ATT-2, Panel Lineup Checklist Unit 3, Rev. 87
3-OI-74/ATT-3, Electrical Lineup Checklist Unit 3, Rev. 88
SRs: 652649, 652431

Section 1R05: Fire Protection

Fire Protection Report, Volume 1, Fire Protection Plan, Units 1/2/3, Rev. 14
Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-519 Torus Area and HPCI Room, Rev. 48
Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-519NW, Rev. 48
Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-519SW, Rev. 48
Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-565, Rev. 48

Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-593, Rev. 48
 Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-621, Rev. 48
 Fire Protection Report, Volume 2, Sections IV, Pre-Plan No. RX3-639, Rev. 48
 Fire Protection Report, Volume 1, Fire Hazards Analysis, Units 1/2/3, Rev. 14

Section 1R08: Inservice Inspection Activities (71111.08G)

Procedures

54-ISI-363-007(AREVA), Remote Underwater In-Vessel Visual Inspection of Reactor Pressure Vessel Internals, Components, and Associated Repairs in Boiling Water Reactors, Rev. 7
 MMDP-10, Controlling Welding, Brazing, And Soldering Processes, Rev. 11
 MMDP-8, Controlling Welding, Brazing, And Soldering (WBS) Materials, Rev. 4
 MMDP-9, Qualification, Certifications of Personnel Performing Welding Processes, Rev. 6
 N-MT-6, TC 11-09, Administration of NDE Procedures for Magnetic Particle Examination, Rev. 6
 NPG-SPP-03.1, Corrective Action Program, Rev. 5
 NPG-SPP-03.1.4, Corrective Action Program Screening and Oversight, Rev. 9
 NPG-SPP-03.1.7, PER Analysis, Actions, Closures and Approvals, Rev. 8
 NPG-SPP-09.7, Corrosion Control Program, Rev. 2
 N-RT-1, Radiographic Examination of Nuclear Power Plant Components, Rev. 28
 N-UT-4, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Rev. 11
 N-UT-76, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, Rev. 7
 N-VT-1, Visual Examination Procedure for ASME Section XI Preservice and Inservice, Rev. 44
 O-TI-140, Monitoring Program for Flow Accelerated Corrosion, Rev. 4
 O-TI-365, Unit 1 Reactor Pressure Vessel Internals Inspection (RPVII), Rev. 3

Corrective Action Documents

PER 541322
 PER 536424
 PER 562226
 PER 533410
 PER 526313
 PER 631474
 PER 465158
 PER 295682
 PER 461544
 PER 278797
 PER 545569
 PER 533699
 SR 638639
 SR 642831

Other

001, Indication Notification Report of Steam Dryer, Rev. 0
 1-SI-4.6G, Inservice Inspection and Risk-Informed Inservice Inspection Program Unit 1, Rev. 26
 Areva Certificate of Personnel Qualification (EVT-1/BWRVIP) (Brown), ID# B7717
 Areva Certificate of Personnel Qualification (EVT-1/BWRVIP) (Telschow), ID# T5817
 Areva Certificate of Personnel Qualification (Vision) (Telschow), ID# T5817
 Densitometer, Serial Number 027605
 Dwg#1-47B452H0158, Mechanical RHR System Pipe Support, Rev. 0

Dwg#1-47E812-1-ISI, ASME Section XI HPCI- Code Class Boundary, Rev. 11
 Dwg#1-FAC-001-036 (CSI), Unit 1 FAC Location Sketch, Main Steam Lines from Manifold HDR to the HP Turbine and Bypass Valve Loop, Rev. 0
 Dwg#1-FAC-006-044 (CSI), Unit 1 FAC Location Sketch, Misc. 8" Drain Header A, B, & C to Condenser 1A, 1B, & 1C, Rev. 0
 Dwg#1-FAC-006-052 (CSI), Unit 1 FAC Location Sketch, OPER Vent Lines from Feedwater Heater A4/B4 & C4 to Condensers A/B & C,, Rev. 0
 Dwg#1-ISI-0091-C, HPCI Weld Locations, Rev. 0
 Dwg#1-ISI-0363-C, RHR Shutdown Support Locations, Rev. 0
 Dwg#HPCI-1-018-4, HPCI System Weld, Rev. 67
 PQR GT-11-0-1
 PQR GT-11-SPEC-1
 Source Certificate for Ir192, Holder# 79637B
 Structural Integrity Certificate of Personnel Qualification (UT) (May), ID# 1908
 Structural Integrity Certificate of Personnel Qualification (Vision) (May), ID# 1908
 Structural Integrity Certificate of Personnel Qualification (Visual) (May), ID# 1908
 TVA Certificate of Personnel Qualification (MT) ((Priestley), ID# 1UPWAOJ7H
 TVA Certificate of Personnel Qualification (RT) (Fox), ID# TDM143XY3
 TVA Certificate of Personnel Qualification (RT) (Melford Sr.), ID# PCJ49PAS1
 TVA Certificate of Personnel Qualification (UT) (Case), ID# 9XUIL0MVC
 TVA Certificate of Personnel Qualification (UT) (Welch), ID# RGV1VT3
 TVA Certificate of Personnel Qualification (Vision) (Case), ID# 9XUIL0MVC
 TVA Certificate of Personnel Qualification (Vision) (Fox), ID# TDM143XY3
 TVA Certificate of Personnel Qualification (Vision) (Ledford), ID# 905506
 TVA Certificate of Personnel Qualification (Vision) (Melford Sr.), ID# PCJ49PAS1
 TVA Certificate of Personnel Qualification (Vision) (Priestley), ID# RGV1VT3
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 TVA Certificate of Personnel Qualification (VT) (Case), ID# 9XUIL0MVC
 TVA Certificate of Personnel Qualification (VT) (Priestley), ID# 1UPWAOJ7H
 TVA Certificate of Personnel Qualification (VT) (Welch), ID# RGV1VT3
 TVA Certificate of Personnel Qualification (Welding) (Dwens), ID# RX4MTJMA0
 TVA Certificate of Personnel Qualification (Welding) (Garne), ID# 89GUHH9BE
 TVA Certificate of Personnel Qualification (Welding) (McCrelen), ID# 3L6CVC05B
 TVA Certificate of Personnel Qualification (Welding) (Pierce), ID# FW1CJ5V48
 TVA Certificate of Personnel Qualification (Welding) (Potts), ID# 27E0204B0
 TVA Certificate of Personnel Qualification (Welding) (Potts), ID# HCQLUFOQD
 TVA Certificate of Personnel Qualification (Welding) (Terry), ID# BFK8QFZ8U
 TVA Certificate of Personnel Qualification (Welding) (Tomkins), ID# Z09MA8T2D
 TVA Certificate of Personnel Qualification (Welding) (Tucker), ID# MWM1KT9SP
 TVA Certificate of Personnel Qualification (Welding) (Whitley), ID# HEF006ZH1
 URS Certificate of Personnel Qualification (UT) (Butler), ID# 23863
 URS Certificate of Personnel Qualification (UT) (Fish), ID# 11401
 URS Certificate of Personnel Qualification (UT) (Fish), ID# 61771
 URS Certificate of Personnel Qualification (Vision) (Butler), ID# 23863
 URS Certificate of Personnel Qualification (Vision) (Fish), ID# 11401
 URS Certificate of Personnel Qualification (Vision) (Fish), ID# 61771
 URS Certificate of Personnel Qualification (VT) (Butler), ID# 23863
 URS Certificate of Personnel Qualification (VT) (Fish), ID# 11401

URS Certificate of Personnel Qualification (VT) (Fish), ID# 61771
WPS, DWPS GT-11-0-1-N, Rev. 2

Section 1R11: Licensed Operator Requalification

Simulator Exercise Guide, (SEG), Rev. 2

Records:

License Reactivation Packages (2 Records Reviewed)
LORP Training Attendance records
Medical Files (16 Records Reviewed)
Remedial Training Records (Various)
Remedial Training Examinations (2 Records Reviewed)
Various condition reports over the last two years related to licensed operator on shift performance
Various closed condition reports that were simulator related

Written Examinations:

2011 RO week 1
2011 SRO week 1
2011 RO week 3

Annual Examination Scenarios:

LOR-EXAM- 26, REV.3
LOR-EXAM- 27, REV.3
LOR-EXAM- 51A, REV. 3
LOR-EXAM- 19, REV. 3
LOR-EXAM- 41, REV. 3
LOR-EXAM- 50a, REV. 3

LOR Practice scenarios:

OPL177.060 Rev 9
OPL177.084 Rev 4
OPL177.093 Rev 1

JPMs:

JPM-70ap- Secure Drywell Sprays
JPM 177TC- Secondary Containment Radiation Alert
JPM 204 U2 -Secure System II from suppression pool cooling
JPM 231ap (U1) -Inhibit ADS
JPM 222 r2- Perform Control Room Transfer of 4KV Unit Board 2B Power Supplies
JPM 234 -Operator 4 Manual Actions 0-SSI-21
JPM254-1-EOI Appendix-7C
JPM 263ap-Spreading Room Smoke Removal
JPM 265ap-Unit 2 Recirc Pump Recovery with manual scram
JPM 266-USST 1B Transformer Tap Changer Auto Checks

Procedures:

NPG-SPP-17.4.1 Exam Security and Exam Database Management Rev. 05, (07-31-2012)
 NPG-SPP-17.8.1 Licensed Operator Requalification Examination Development and Implementation, Rev. 07, (05-31-2012)
 NPG-SPP-17.8.2 Job Performance Measures Development, Administration, and Evaluation, Rev. 02, (04-04-2012)
 NPG-SPP-17.8.3 Simulator Exercise Guide Development and Revision, Rev. 02, (03-30-2012)
 NPG-SPP-17.8.4 Conduct of Simulator Operations, Rev. 0, (12-27-2011)
 TRN-12 Simulator Regulatory Requirements, Rev. 11, (11-02-2011)

Simulator Static and Normal Tests:

100% Steady State Test, Revision 11
 82% Steady State Test, Revision 11
 46% Steady State Test, Revision 11
 Unit 3 Simulator Normal Testing of GOI's Revision 11

Simulator Transient Tests:

Transient Test #1, Manual Scram, Revision 11
 Transient Test # 4, Simultaneous Trip of All Recirculation Pumps, Revision 11
 Transient Test # 6, Turbine Trip < 30% Power, Revision 11
 Transient Test # 7, Maximum Rate Power Ramp, Revision 11

Simulator Malfunction Tests:

RP06-Auto Scram Channel Failure, Revision 11
 TC08-Control Valve Position Unit Failure, Revision 11
 TC10-EHC Pressure Transducer Failure, Revision 11
 ED27-Loss of Power to an ECCS 250V RMOV Board Breaker Failure, Revision 11

PERs:

PER 595296 Operation Missed Technical Specification Call
 PER 566196 Develop Case Study for Drain Down Event
 PER 558521 Shift Manning in the U3Control Room Inadequate for Start up
 PER 245312 Reactivity Management Plan (RCP) requires improvement

Standards:

ANSI/ANS-3.5-1985, American National Standard Nuclear Power Plant Simulators for Use In Operator Training and Examination
 ANSI/ANS-3.4-1983, Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants

Other Documents:

Self Assessment BFN-TRN-12-010 Operations Training Department IP71111.11B Inspection Preparations, (July 7-September 10, 2012)
 Snapshot self-Assessment Report BFN-OPS-S-004 February 10-14, 2012
 Reviewed three LERs for Unit 3 and 1 for Unit 1

Section 1R12: Maintenance Effectiveness

0-SR-3.7.3.4, Control Bay Habitability Zone Pressurization Test
 Air Conditioning (a)(1) plan
 Crevs (a)(1) plan
 MCI-0-000-PCK001, Generic Maintenance Instructions for Valve Packing, Rev.29
 NETP-117, Valve Stem Packing Enhancement Program, Rev. 0
 NPG-SPP-03.4, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting 10CFR50.65
 PER 236646, A Fleet Valve Stem Packing Program Should be Implemented
 PER 244202, BFN-3-VTV-10-502 Blown Packing
 PER 252382, MR (a)(1) Plan due to Trend in Plant Shutdown Events Induced by Valve Packing HU Events
 PER 329005, CREVS in (a)(1) status
 PER 423569, System 31 Maintenance Rule Performance Criteria Exceeded
 PER 473637, 1-FCV-68-79 Drywell Packing Leak
 PER 533052, MSIV LLRT Failure due to Valve Packing Blowby
 PER 565652, System 31 (a)(1) Plan
 PER 567503, (a)(1) Plan Interim Performance Criteria Exceeded
 PER 614107, Evaluate need for additional System 031 PMs
 PER 652791, Excess Tripping of AC Units
 TVA NPG Quick Human Error Analysis Tool, PER 244202, 8/12/2010
 TVA Nuclear Power Group BFN Engineering Support Morning Status, Degraded Conditions/Non-Conforming

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

BFN Plan of the Day, 10/25-26/2012
 U1R9 Risk Plan, Inventory Control During An OPDRV Activity (TI-106)
 Operator Aid (Drawing) for Unit 1 RHR tie to 1B Recirculation Pump Discharge piping
 NRC Staff Position on Dispositioning Boiling-Water Reactor Licensee Noncompliance with Technical Specification Requirements During Operations with a Potential for Draining the Reactor Vessel
 Enforcement Guidance Memorandum (EGM) 11-003, "Enforcement Guidance Memorandum on Dispositioning Boiling-Water Reactor Licensee Noncompliance with Technical Specification Containment Requirements during Operations with a Potential for Draining the Reactor Vessel"
 Procedure 2-POI-200.5, Operations with Potential for Draining the Reactor Vessel/Cavity, Rev 14
 NPG-SPP-09.11.2, Equipment Out of Service (EOOS) Management, Rev. 5
 NPG-SPP-07.2.11, Shutdown Risk Management, Rev. 2, dated 10/4/2011
 ORAM Outage Safety Assessment for November 1, 2012 Orange risk
 NRC Regulatory Issues Summary 2012-11,
 NPG Daily Outage Report, U1R9, 10/25-26/2012
 WO 114056943
 2-SR-3.4.2.1, Jet Pump Mismatch and Operability, Rev. 34
 2-SR-3.4.1(SLO) Reactor Recirculation System Single Loop Operation, Rev. 09
 U2 RCP 121130-000, Reactivity Maneuver Plan U2 Single Loop Operation (SLO)

EOOS Operator's Risk Worksheet, 12/13/2012
 NPG-SPP-09.11.1, Equipment Out of Service (EOOS) Management, Rev. 05

Section 1R15: Operability Evaluations

0-AOI-100-3, Flood Above Elevation 558', Rev. 35
 0-OI-67, Emergency Equipment Cooling Water System, Rev. 96
 Calculation MDQ0067910008, Flow Requirements of EECW Fed Components, Rev. 16
 Calculation MDQ0023870149, RHRSW Pump Compartment Sump and Sump Capacity, Rev. 10
 Design Criteria BFN-50-7067, Emergency Equipment Cooling Water System
 FSAR Section 1.2.72, Probable Maximum Flood, BFN-21
 FSAR Section 2.4, Hydrology, Water Quality, and Aquatic Biology, BFN-19
 FSAR Section 10.10, Emergency Equipment Cooling Water (EECW) System, BFN-22
 FSAR Section 12.2.7.1.2, Principle Structures and Foundations, Personnel Access Doors, BFN-24
 FSAR Appendix 2.4A, Browns Ferry Nuclear Plant Maximum Possible Flood, BFN-24
 PER 617833, B3 RHRSW/EECW Pump Has Shaft Seal Leak
 PER 617840, D EECW South Header Strainer Leaking
 SR 622301, Potential Non-Conservative Assumptions in Calculation for Leakage into RHRSW Pump Rooms
 Technical Specification and Basis 3.7.2 Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS), Amendment 235 and Rev. 0 respectively
 0-TI-403, Appendix A, Determination of Common Cause Failure for Emergency Diesel Generators, Rev. 0, dated 10/04/12
 Design Criteria BFN-50-7082, Standby Diesel Generator
 Drawing 3-45E767-5, Wiring Diagram Diesel Generators Schematic Diagram, Rev. 26
 Engine Systems Inc. Bill of Material, TVA-Browns Ferry/EDG Governor Upgrade, ESI IWO 8001206 & 8002031, Rev. 5
 FSAR Section 8.5, Standby AC Power Supply and Distribution, BFN-24
 OPDP-1, Conduct of Operations, Rev. 24
 PER 617890, 3EB EDG Relay Failed During Dynamic Testing
 PER 619824, Reconfigured Relays with Incorrect Part Numbers Installed in 3EB Governor Upgrade Modification
 PER 619972, Failure to Document Critical Thinking per OPDP-1
 PER 621030, WO Needed to Investigate and Resolve Configuration of Relays Installed in 3EC EDG Governor
 PER 621079, Timeliness of PDO for 3C Diesel Generator
 SR 628201, PER 619824 Requires Re-Screening
 SR 628216, Improper Classification of PER 617890
 SR 628627, Repeat Failure of a CC2 Component
 Technical Specifications Task Force (TSTF) – 531, Revision of Specification 3.8.1, Required Actions B.3.1 and B.3.2, Rev. 0
 Unit 3 Technical Specification and Basis 3.8.1 AC Sources - Operating, Amendment 266
 WO 114009014, DG 3C Resolution of Relay Configurations
 Drawing 0-46W401-10, Architectural Plans EL 519 and 565, Rev. 0
 Drawing 0-34N303, Watertight Personnel Access Doors, Rev. 0
 Drawing 1-47E852-1, Flow Diagram Floor & Dirty Radwaste Drainage, Rev. 26
 PER 623264, Broken Weld on Door Frame BFN-0-DOOR-260-230A

WO 07-717762-00, Repair Door Seal
 WO 114020199, Repair Broken Weld on Door Frame BFN-0-DOOR-260-230A
 Design Criteria BFN-50-7073, High Pressure Coolant Injection System, Rev. 22
 Flowserve Design, Seismic, and Weak-Link Analysis, RAL-2634, Size 10 Class 900 Carbon Steel Double Disc Gate Valve, Rev. 2
 Flowserve Drawing , Anchor/Darling, BW/IP, Durco and Valtek Valves, 10" – 900 lb. Double Disc Gate Valve Weld Ends, Carbon Steel, Body Drain Pipe with Cap, Smart Stem and AdvanSeal with Limitorque SMB-2-80 Actuator, Rev. B
 Flowserve Instruction Manual FCD ADENIM0003-00, Anchor Darling Double-Disc Gate Valves
 FSAR Section 6.4.1, High Pressure Coolant Injection System, BFN-24
 FSAR Section 7.4.3.2, High Pressure Coolant Injection System (HPCI) Control and Instrumentation, BFN-24
 PER 627529, 1-FCV-73-2 As-Found Leak Rate was 600 scfh with an Admin Limit of 30 scfh
 PER 639155, Internal Damage Found in 1-FCV-73-2
 PER 638761, Internal Inspection of 1-FCV-73-3
 SR 642135, Perform Inspection on 2-FCV-73-2 Wedge Pin During U2R17 Outage
 Technical Specification and Basis 3.5.1 EECS - Operating, Amendment 269 and Rev. 53 respectively
 Vendor Technical Document BFN-A391-0340, Instruction Manual for Anchor Darling 10" – 900 Double Disc Gate Valve, Rev. 5
 WO 02-011512-001, Remove/Replace 1-FCV-73-2 with New Design
 WO 08-711832-000, 1-FCV-73-3 Repaired Due to Wrong Pinion/Worm Gear
 WO 110937504, 1-FCV-73-16 Repaired Due to Pressure Locking
 WO 00-003350-000, 2-FCV-73-2 Pin Sheared During MOVATS
 WO 112075251, 2-FCV-73-3 Stem Replaced
 WO 110811072, 2-FCV-73-16 Reworked due to Seat Leakage
 WO 112330027, 3-FCV-73-16 Replaced due to Seat Leakage
 0-OI-23, Residual Heat Removal Service Water System, Rev. 92
 Calculation MDQ0023890078, Pump Performance Analysis for New RHRSW Compartment Sump Pumps, Rev. 3
 Design Criteria BFN-50-7023, Residual Heat Removal Service Water System
 PER 618735, Removed Scrap Wire From RHRSW Pump Room Sumps
 PER 623106, Potential Non-Conservative Assumptions in Calculation for Leakage into RHRSW Pump Room Sump
 SR 622301, Potential Non-Conservative Assumptions in Calculation for Leakage into RHRSW Pump Rooms
 FSAR SBGT
 R14 981211 106 TVA Alternate Source Term Calculation Input Parameters
 R 92 960718 850 TVA Alternate Source Term Calculations
 PER 590208 Past Operability Determination for SBGT C relative humidity heater inoperable
 PER 604350 Breaker for SBGT C relative humidity heater inoperable
 LER 50-259/2012-008 SBGT Train C inoperable longer than Technical Specification allowable time
 WO 113759132 Standby Gas Treatment Train C filter heating element lost power

Section 1R18: Plant Modifications

NPG-SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 9
 NPG-SPP-6.9.3, Post-Modification Testing, Rev. 3
 DCN 70488, Replace Flowserve Gate Valve with Crane-Kalsi Sentinel Gate Valve, Rev. B
 WO 113086731, Implement DCN 70488 to Replace BFN-1-FCV-0016
 WO 113657859, Perform MOVATS to Support WO 113086731 for Valve Replacement
 WO 114009989, Perform Pre-fab Welding on New HPCI Valve
 UFSAR, Section 6.4.1 High Pressure Coolant Injection System, BFN-24
 DWG DCA No. 70488-119, CD05897 RB, Crane Bolted Bonnet Gate Valve, Rev. Orig.
 MCI-0-000-GTV001, Generic Maintenance Instructions for Gate Valves, Rev. 28
 MCI-0-000-PCK001, Generic Maintenance Instructions for Valve Packing, Rev. 29

Section 1R19: Post-Maintenance Testing

WO 111436112 - Replace 0-HS-023-0015A
 WO 113195620 – Abandon RHRSW pp B2 local controls
 0-TI-403, Appendix A, Determination of Common Cause Failure for Emergency Diesel Generators, Rev. 0, dated 10/04/12
 DCN 69532 Stage 7 Continuity Checks
 Design Criteria BFN-50-7082, Standby Diesel Generator
 Drawing 3-45E767-5, Wiring Diagram Diesel Generators Schematic Diagram, Rev. 26
 Engine Systems Inc. Bill of Material, TVA-Browns Ferry/EDG Governor Upgrade, ESI IWO 8001206 & 8002031, Rev. 5
 FSAR Section 8.5, Standby AC Power Supply and Distribution, BFN-24
 PER 617890, 3EB EDG Relay Failed During Dynamic Testing
 PER 619824, Reconfigured Relays with Incorrect Part Numbers Installed in 3EB Governor Upgrade Modification
 PER 621030, WO Needed to Investigate and Resolve Configuration of Relays Installed in 3EC EDG Governor
 PER 621079, Timeliness of PDO for 3C Diesel Generator
 PER 629871, Evaluation of Standby diesel Generator 3B and 3C Relay Issues following Implementation of DCN 69532 in Response to NRC Concerns
 PMTI-69532-STG007, 3C Emergency Diesel Generator Governor Control Upgrade, Rev. 2
 PER 629081, 3C Diesel PMT Not Ready to Work Due to Inadequate Planned PMT
 PER 642810, During Performance of PMT IAW WO 114009014, 3C Start Relay BFN-3-RLY-82-3C2
 Technical Specifications Task Force (TSTF) – 531, Revision of Specification 3.8.1, Required Actions B.3.1 and B.3.2, Rev. 0
 Unit 3 Technical Specification and Basis 3.8.1 AC Sources - Operating, Amendment 266
 WO 111549226, Attachment 1, Diesel Generator 2301A and EGB-13P Governor Setup & Tuning Instruction
 WO 114009014, DG 3C Resolution of Relay Configurations
 WO 114114698, 3-SR-3.8.1.1(3C) Diesel Generator 3C Monthly Operability Test
 1-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations, Rev. 19
 1-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring, Rev. 8
 Drawing 1-47E811-1, Flow Diagram Residual Heat Removal System, Rev. 37
 OPDP-1, Conduct of Operations, Rev. 24
 SR 633312, Unit 1 Suppression Pool Water Level Has Dropped 2.2” Between 10/26 and 10/30

SR 634421, NRC Has Identified the RHR Flow Drawing Does Not Show Two Isolation Valves
 WO 113622372, 1-SR-3.4.9.1(1)
 3-SR-3.5.1.6 (RHR I), Quarterly RHR System Rated Flow Test Loop I, Rev. 42
 3-SR-3.5.1.6 (RHR I-COMP), RHR Loop I Comprehensive Pump Test, Rev. 06
 SR 653678, 3C RHR pump discharge pressure indicator
 EPI-0-000-BKR015, 4KV Wyle/Siemens Horizontal Vacuum Circuit Breaker (Type-3AF) and
 Compartment Maintenance
 1-SR-3.5.1.8, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at 150
 psig Reactor Pressure, Rev. 11
 WO 113195499, 1-SR-3.5.1.8
 WO 113657859, Perform NOVATS to support WO 113086731 for Valve Replacement
 WO 113657859, Attachment to WO Task 10, Rev. 3
 ECI-0-000-MOV009, Testing of Motor Operated Valves Using MOVATS Universal Diagnostic
 System (UDS) and Viper 20, Rev. 28
 UFSAR, Section 6.4.1 High Pressure Coolant Injection System, BFN-24
 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test, Rev. 21
 WO 113195490, HPCI Comprehensive Pump Test
 MSI-1-073-GOV001, High Pressure Coolant Injection (HPCI) Turbine Overspeed Trip Test,
 Rev. 9
 0-TI-383, Evaluation of Test Results for the ASME OM Code Inservice Testing Program, Rev. 1
 ASME OM Code IST Test Results Evaluation UNID: 12-1-IST-073-473, dated 11/27/2012, SR
 648090
 PER 648108, 73-16 Stroke Times
 1-SR-3.3.3.1.4(G), Verification of Remote Position Indicators for HPCI System Valves, Rev. 2
 WO 113183659, 1-SR-3.3.3.1.4(G)
 PER 649588, Drain Valve Double Lit
 WO 114164455, Drain Valve Double Lit
 1-SR-3.6.1.3.5(HPCI), HPCI System Motor Operated Valve Operability, Rev. 9
 WO 113767670, 1-SR-3.6.1.3.5(HPCI),

Section 1R20: Refueling and Other Outage Activities

1-GOI-100-3A, Refueling Operations (RX Vessel Disassembly and Floodup), Rev. 23
 1-GOI-100-3B, Refueling Operations, (Reactor Cavity Letdown and Vessel Re-Assembly),
 Rev.21
 0-GOI-100-3A, Refueling Operations (In-Vessel Operations), Rev. 55
 0-GOI-100-3B, Operations in Spent Fuel Storage Pool Only, Rev. 49
 0-GOI-100-3C, Fuel Movement Operations During Refueling, Rev. 68
 NPG-SPP-05.8, Special Nuclear Material Control, Rev. 3
 Fuel Assembly Transfer Form, BFN Nuclear Plant, Transfer Operation No. BFN-1-129
 PER 635794, Fuel Move Orientation Error on 11/01/2012
 PER 636511, FME on Bundle JYP303 at position 31-06
 PER 637790, Fuel Assembly JYE 347 was found improperly seated during core verification
 PER 651334, Reactor Pressure Vessel inner O-ring leak ODMI
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 Tagout: 1-TO-2012-003, Clearance 1-073-0021, HPCI Pump Injection Valve
 Tagout: 1-TO-2012-003, Clearance 1-074-0018B, RHR System I Drywell Spray Outboard Valve

Section 1R22: Surveillance Testing

Diesel Generator C Emergency Unit 1 Load Acceptance Test; 0-SR-3.8.1.9(C), Rev 7
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 3-47E852-2, Flow Diagram for Clean Radwaste & Decontamination Drainage, Rev. 36
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 1-SR-3.5.1.8, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at 150
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Section 1EP4: Emergency Action Level and Emergency Plan Changes**Change Packages**

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 Emergencies," Revision 37
 TVA Radiological Emergency Plan, Revision 97 and 98

Section 4OA2: Identification and Resolution of Problems

PER 618132, Relay 71X-85-45H3 Failure
 ACE PER 618132, Unit 2 Half Scram Due to Failed HFA Relay
 PER 571080, U3 Received 'A' Channel Half Scram due to failed relay
 ACE PER 571080, Unit 3 Half Scram due to Failed HFA Relay
 PER 621034, Unit 2 RPS Relay Failure
 PER 624280, Lower Tier B Level 571080 on Relays had Incorrect Cause and Extent of
 Condition
 PER 626992, 1-RLY-064-16A-K78 is normally energized, safety related with Clear Lexan Spool
 PER 630742, Discrepancy in NPG Procedures
 PER 644806, HFA Relay Premature Failure
 PER 654390, As part of Extent of Condition, Replace Relay
 PER 659897, NRC Identified Request
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 DWG 2-730E915RF, Sheet 11, Reactor Protection System, Rev. 14
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 NPG-SPP-09.18, Integrated Equipment Reliability Program, Rev. 3
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 Integrated Trend Report Q3FY12, dated 8/13/2012
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 PIDP-20, Corrective Action Program Lower Level Metrics, Rev. 2
 Site Cap Health Monitor, 12/2012
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 NPG-SPP-03.14, Corrective Action Program Screening and Oversight, Rev. 10
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 1-47E225-103, Unit 1 Harsh Environment Data, EL '519, Rev 002, 3/18/2003
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 R06121212978, EQIR No. BFNEQ121004, Subject EQDP BFN0EQ-CABL-034, 11/28/2012
 R06121024805, EQIR No. BFNEQ09809, Subject EQDP BFN0EQ-CABL-034, 8/18/2009
 R06121026825, EQIR No. BFNEQ12954, Subject EQDP BFN0EQ-CSC-001, 10/24/2012

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 R06121026823, EQIR No. BFNEQ12898, Subject EQDP BFN0EQ-IZS-004, 10/24/2012
 PER 283220, WO 111615866 Replaced 2-PS-075-0007 with a Model Number Different from
 that Previously Installed
 PER 238931, EQIRs in Violation of SPP-9.2
 PER 624137 NPG-SPP-09.3 Does Not Require Timely DCN Closure
 PER 559420 EQ Equipment Installed without Supporting EQ Binder
 PER 575444 Drawing discrepancy on 1-47E225-103 and 2-47E225-103

EQ Components Selected for Walkdown

BFN-2-CSC-023-0048 Conduit Seal for 2-FT-023-0048
 BFN-2-CSC-074-0064 Conduit Seal for 2-FT-074-0064
 BFN-2-FT-023-0048 RHR HTX B SW Flow
 BFN-2-FT-074-0064 RHR Loop 2 Flow
 BFN-2-HS-074-0098B RHR Pump B Suction Crosstie Valve
 BFN-2-HS-074-0099B RHR Pump D Suction Crosstie Valve
 BFN 2-MTR-074-0028 RHR Pump 2B Motor
 BFN 2-PS-074-0031A RHR Pump B Discharge Pressure
 BFN 2-PS-074-0031B RHR Pump B Discharge Pressure
 BFN 2-MTR-073-0026 HPCI Suppression Pool Inboard Suction Valve
 BFN-2-MTR-073-0027 HPCI Suppression Pool Outboard Suction Valve
 BFN-2-MVOP-073-0026 HPCI Suppression Pool Inboard Suction Valve Operator
 BFN-2-MVOP-073-0027 HPCI Suppression Pool Outboard Suction Valve Operator
 BFN-2-CSC-075-021 Conduit Seal at 2-FT-75-21
 BFN-2-CSC-075-0057A Conduit Seal Connector for Limit Switch on FCV-75-57
 BFN-2-FSV-075-0057 Solenoid Valve for Drain Pump A Bypass Line
 BFN-2-FT-075-0021 System 1 Flow CS Flow SQ RT
 BFN-2-MTR-075-0005 Core Spray Pump A Motor and Core Spray Pump C Motor
 BFN-1-CSC-023-0048 Conduit Seal for 1-FT-023-0048
 BFN-1-CSC-074-0064 Conduit Seal for 1-FT-074-0064
 BFN-1-FT-023-0048 RHR HTX B SW Flow
 BFN-1-FT-074-0064 RHR Loop 2 Flow
 BFN-1-HS-074-0098B RHR Pump B Suction Crosstie Valve
 BFN-1-HS-074-0099B RHR Pump D Suction Crosstie Valve
 BFN 1-MTR-074-0028 RHR Pump 2B Motor
 BFN 1-PS-074-0031A RHR Pump B Discharge Pressure
 BFN 1-PS-074-0031B RHR Pump B Discharge Pressure
 BFN 1-MTR-073-0026 HPCI Suppression Pool Inboard Suction Valve
 BFN-1-MTR-073-0027 HPCI Suppression Pool Outboard Suction Valve
 BFN-1-MVOP-073-0026 HPCI Suppression Pool Inboard Suction Valve Operator
 BFN-1-MVOP-073-0027 HPCI Suppression Pool Outboard Suction Valve Operator
 BFN-1-CSC-075-0021 Core Spray System Flow
 BFN-1-CSC-075-0057A Pressure Suppression High Level Control

PERs Generated as a Result of Inspection

SR 659324 Maximo Indicates Incorrect Room Location for Valve Operator and Valve Motor
 PER 658970 Second Party Verification Not Performed for Required SPV Fields in Maximo
 PER 658469 Fields in Maximo Not Having Required Second Party Verification

PER 659324 Incorrect Room Reference in Maximo for 1-MVOP-73-26 and 1-MTR-73-26
 PER 660812 Maximo indicates incorrect Room location for valve operator and valve motor.

Section 40A3: Event Follow-up

FSAR Unit 1

R14 981211 106 TVA Alternate Source Term Calculation Input Parameters

R 92 960718 850 TVA Alternate Source Term Calculations

PER 590208 Past Operability Determination for SGBT C relative humidity heater inoperable

PER 604350 Breaker for SGBT C relative humidity heater inoperable

LER 50-259/2012-008 SGBT Train C inoperable longer than Technical Specification allowable time

WO 113759132 Standby Gas Treatment Train C filter heating element lost power

Root Cause Analysis (RCA) SGBT Train C inoperable

Section 40A5: Other Activities

0-AOI-100-3, Flood Above Elevation 558', Rev. 35

CTP-FWD-100, Flood Protection Walkdowns NEI 12-07, Rev. 0

FSAR Appendix 2.4A, Browns Ferry Nuclear Plant Maximum Possible Flood, BFN-24

MPI-0-000-INS001, Inspection of flood Protection Devices, Rev. 12

NEI 12-07, Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features, Rev. 0-A

NRR Japan Lessons-Learned Project Directorate Letter dated 5/31/2012, Endorsement of Nuclear Energy Institute (NEI) 12-07, "Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features"

PER 469640, Aggregate Impact of RHRSW Pump Room Watertight Door Degradations

PER 647926, NTTF 2.3 Flooding Walkdown Small Available Physical Margin

PER 568642, RHRSW Pump Room Watertight Door Degradations

PER 589442, NRC Concerns of Gaps in the NTTF-2.3 Flood Walkdown Scope

Adverse Employment Action Procedure

TVA Fleet ECP Reports

Corrective Action Documents

SR626352 Seismic walk down of 3A Diesel Generator anchorage – oil leak

SR626335 Seismic walk down – RHR pipe support anchor loose bolts

Procedures:

CTP-FWD-100, Flood protection Walkdowns NEI 12-07, Rev. 0001

Calculations:

R14 120309119, Water Chiller 3A, Concrete Anchor Edge Distance Evaluation, Control Bay

R14 090211101, RHR System and Core Spray System Pumps Anchor Bolts

Drawings

41N734, Miscellaneous Foundations Elevated 519 Outline and Reinforcement, Reactor Building Units – 1, 2, 3

48N1027, Miscellaneous Steel Pipe Anchor Framing, Reactor Building Units – 1, 3

0-45N328, Electrical Equipment 480 V Diesel Auxiliary BDS A & B Outline and general Equipment

3-45N337-4, Electrical Equipment 480 V Diesel Auxiliary BDS A & B Outline and general Equipment - Unit 3
48N888, Miscellaneous Steel Frames, Covers, Grating and Stairs, EL 583.5, Diesel Generator Building
48N897-3, 5, 8, Miscellaneous Steel Frames, Covers, Grating and Stairs, EL 583.5, Diesel Generator Building - Unit 3
O-45N329, Electrical Equipment Diesel Generator Room A, B, C and D, Units 1, 2 Outline and general Layout
3-45N337-5, Electrical Equipment Diesel Generator Room 3A, 3B, 3C, 3D Outline and general Arrangement, Diesel Generator Building – Unit 3
1-478450-267, Mechanical RHRSW System Pipe Support
O-17W925-1, 3, Mechanical Heating and Ventilating Diesel generator Building Units 1-3
48N897-8, Mechanical Pumping Station and Water Treatment-Piping and Equipment Water Supply Units 1-3
48NI233, Miscellaneous Steel H & V Equipment Anchorage Reactor Building Control Bay – Unit 3
31N203, Concrete Pumping Station Outline-Sheet 1 Condenser Water Supply Units 1-3
17W925-2, Mechanical Heating, ventilating and Air Conditioning, Diesel Generator Building Unit 3
0-41N577-2, Chiller Enclosure Concrete Outline, Diesel generator Building Units 1, 2
0-47E865-8, Flow Diagram Heating, Ventilating and A/C Airflow, Diesel generator Building Units-1, 2
3-41N590-1, Concrete Floors and Walls Outline, Diesel Generator Building Unit 3

Other Documents

EPRI 1025286 "Seismic Walkdown Guidance for Resolution of Fukushima Near-Term Task Force Recommendation 2.3
Browns Ferry Seismic Walkdown Equipment List (SWEL)
Area Walk-By Checklist and Seismic Walkdown Checklist
Personnel Qualification Records

LIST OF ACRONYMS

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