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

July 26, 2017

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT – NRC INTEGRATED INSPECTION REPORT 05000259/2017002, 05000260/2017002, AND 05000296/2017002

Dear Mr. Shea:

On June 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. On July 14, 2017, the NRC inspectors discussed the results of this inspection with Mr. Steve Bono and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented four findings which were determined to be of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. Because of their very low safety significance, the NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy. If you contest any of the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Browns Ferry Nuclear Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II, and the NRC Resident Inspector at Browns Ferry Nuclear Plant.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Alan Blamey, 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 IR 05000259/2017002,
05000260/2017002 and 05000296/2017002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, and 50-296
License Nos.:	DPR-33, DPR-52, and DPR-68
Report No.:	05000259/2017002, 05000260/2017002, and 05000296/2017002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Athens, AL 35611
Dates:	April 1, 2017 through June 30, 2017
Inspectors:	D. Dumbacher, Senior Resident Inspector T. Stephen, Resident Inspector A. Ruh, Resident Inspector J. Dolecki, Acting Resident Inspector
Approved by:	Alan Blamey, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY

05000259/2017002, 05000260/2017002, 05000296/2017002; 04/01/2017- 06/30/2017; Browns Ferry Nuclear Plant, Units 1, 2, and 3; Fire Protection, Flood Protection Measures, Surveillance Testing, Follow-up of Events and Notices of Enforcement Discretion

The report covered a three-month period of inspection by resident inspectors. Four findings were identified. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP) dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross-Cutting Areas" dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6. Documents reviewed, which have not been identified in the Report Details, are listed in the Attachment.

Cornerstone: Initiating Events

- Green. A self-revealing non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was reviewed for the licensee's failure to establish measures to assure that corrective action was taken to preclude repetition of a significant condition adverse to quality (SCAQ). The licensee failed to correct electronic noise problems with the scram reset switch which led to a March 29, 2017, reactor scram. As an immediate corrective action, the licensee initiated more rigorous tests to identify noise vulnerabilities on Intermediate Range and Source Range Monitors. The licensee entered this issue into their corrective action program as Condition Report (CR) 1278595.

  This performance deficiency was more-than-minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective in that the licensee failed to implement corrective actions to address Intermediate Range Monitor (IRM) spiking following the May 24, 2012, reactor scram. The finding was determined to be Green because it did not involve the loss of mitigation equipment. The inspectors determined that the finding had a cross-cutting aspect of Challenge the Unknown (H.11) within the cross-cutting area of Human Performance because the licensee failed to resolve the unknown noise paths to ensure that scram vulnerabilities were corrected. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. An NRC-identified non-cited violation of 10 CFR 50.48(c) and NFPA 805, Section 2.4.2.4 was identified for the licensee's failure to perform an adequate engineering analysis to determine the effects of fire on the ability to achieve the nuclear safety performance criteria. Specifically, the licensee's fire risk evaluation (FRE) of the effects of fire on the Emergency Equipment Cooling Water (EECW) strainers did not have an adequate basis. As an immediate corrective action, the licensee performed plant-specific analyses to determine the effects of fire on the functionality of EECW strainers and EECW system. The violation was entered into the licensee's corrective action program as CR 1263434.
  The performance deficiency was determined to be more-than-minor because it was associated with the protection against external factors attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective in that failure to adequately

analyze the effects of fire damaged cables for the EECW strainers and backwash valves impacted the objective of ensuring the reliability of the EECW system during a fire. This finding was determined to be Green because the finding did not affect the ability to reach and maintain a stable plant condition within the first 24 hours of a fire event. The inspectors determined that the finding had a cross-cutting aspect of Avoid Complacency (H.12) within the cross-cutting area of Human Performance because the licensee did not recognize that historical assumptions about long-term strainer functionality could contain mistakes and latent issues during development of the nuclear safety capability analysis. (Section 1R05)

- Green. An NRC-identified non-cited violation of 10 CFR 50, Appendix B, Criterion III was identified for the licensee's failure to verify the adequacy of the Unit 1 and 2 diesel building emergency drain pipe to mitigate a postulated internal flood. Specifically, the licensee's design review contained non-conservative assumptions. As an immediate corrective action, the licensee reevaluated the potential water accumulation and concluded the diesel generators were still protected. The violation was entered into the licensee's corrective action program as CR 1303737.

  The performance deficiency was more-than-minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, non-conservative assumptions in calculation MDQ00004020110008 resulted in inaccurate conclusions about the capacity of the drain and the resulting water accumulation in the building. The finding was determined to be Green because it represented a deficiency affecting the design of the drain piping, but it maintained its functionality. Functionality was preserved because additional evaluation showed that the resulting water accumulation would not affect any safety related equipment. No cross-cutting aspect was assigned because it was not considered to be reflective of current licensee performance because the performance deficiency occurred more than three years ago. (Section 1R06)

- Green. An NRC-identified non-cited violation of 10 CFR Part 50, Appendix B, Criterion III was identified for the licensee's failure to correctly translate the design basis of the EECW system into technical instruction 0-TI-579(EECW). The effects of instrument uncertainty and diesel frequency variations were not considered when establishing the minimum allowed inservice test low alert pump flow limits. As an immediate corrective action, the licensee evaluated the operability of the EECW pump and initiated corrective action to make changes to the test criteria and/or the system design analysis. The violation was entered into the licensee's corrective action program as CR 1288208.

  The performance deficiency was more-than-minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective in that there was a reasonable doubt on the operability of the B3 EECW pump since portions of the adjusted pump curve would be below the minimum pump curve established in the design basis calculation. Additionally, there was a significant reduction in available margin for the pump under design basis conditions. The finding was determined to be Green because the finding was a deficiency affecting the design of a mitigating system, but the pump maintained its operability. The inspectors determined that the finding had a cross-cutting aspect of Human Performance (H.6) within the cross-cutting area of Design Margins because engineers did not demonstrate the characteristic of ensuring that design margins were guarded and changed only through a systematic and rigorous process. (Section 1R22)

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

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at 100 percent rated thermal power (RTP) except for one unplanned downpower on April 6, 2017, due to condenser waterbox fouling and three planned downpowers for condenser water box cleaning and control rod sequence exchanges.

Unit 2 operated at 100 percent RTP except for one unplanned downpower to 40 percent caused by a trip of 2A Recirculation pump on June 28, 2017, and three planned downpowers for control rod sequence exchanges.

Unit 3 operated at 100 percent RTP except for three planned downpowers for condenser water box cleaning.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors focused on identification of discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down control system components, and determined whether selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program (CAP). The inspectors completed five Equipment Alignment Partial Walkdown samples.

- Unit 3, Hardened containment vent system
- Unit 3, 3D Emergency Diesel Generator (EDG), while 3C EDG was unavailable due to planned 4 year preventative maintenance
- Unit 2, High Pressure Coolant Injection (HPCI) while Reactor Core Isolation Cooling (RCIC) system was out of service
- Unit 1, 2 & 3 EECW system supply and discharge piping
- Unit 3 Standby Liquid Control system with a focus on boric acid concentration, pump and tank heater controls.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Fire Protection Tours

a. Inspection Scope

The inspectors reviewed licensee procedures for transient combustibles and fire protection impairments, and conducted a walkdown of fire areas (FA) or selected compartments of larger fire areas as listed below. These FAs or compartments 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 condition of fire protection features or measures. The inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedures. The inspectors reviewed applicable portions of the Fire Protection Requirements Manual (FPRM) 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 Walkdown inspection samples, as defined in Inspection Procedure 71111.05.

- Compartment 16-O, Auxiliary Instrument Room for Unit 3
- Compartment 16-N, Communications Room, Communications Battery Room, Communications Battery Board Room, and Computer Room
- Fire Area 4, Electric Board Room 1B
- Fire Area 20, Units 1 and 2 Diesel Generator Building
- Compartment 16-A, Unit 1 Cable Spreading Room

b. Findings

Introduction: The NRC identified a Green NCV of 10 CFR 50.48(c) and NFPA 805, Section 2.4.2.4 for the licensee's failure to perform an adequate engineering analysis to determine the effects of fire on the ability to achieve the nuclear safety performance criteria. The licensee's fire risk evaluation (FRE) of the effects of fire on the EECW strainers did not have an adequate basis.

Description: The licensee's nuclear safety capability analysis (NSCA) identified that postulated fires in 17 of 45 fire areas could result in the loss of power to credited EECW strainers and backwash valves. During normal system operations, these strainers are continuously rotated by motors and the backwash valves are opened by motor operators. Strainer rotation allows for continual break up of large debris and the backwash valve opens a flushing path to clean the strainers. In the event that fire results in a loss of power, the strainer would be stationary and the backwash valve could be failed in the closed position. Consequently, debris build-up over time could affect EECW flow to components such as the diesel generators and Residual Heat Removal (RHR) and Core Spray (CS) room coolers. Loss of these components, credited in the analysis, was identified as a variance from deterministic requirements (VFDR). The licensee used a FRE per NFPA 805 section 4.2.4.2 to demonstrate that the increased risk associated with the VFDR was acceptable. The licensee determined that the loss of power was acceptable because the EECW system was designed to operate for 48 hours without backwashing the strainers. Based on this conclusion, no recovery actions to manually rotate and backwash the strainers were included in the fire safe shutdown procedures. Inspectors reviewed the FRE and identified several concerns: 1) the supporting analyses were based on tests performed with conditions that were not representative of the Browns Ferry fire condition due to differences in equipment, configuration, and operation, 2) the evaluation did not discuss the impact of loss of strainer rotation, and 3) the site's operational history included instances where strainers became fouled in less than 48 hours. The licensee subsequently performed functionality evaluations and determined that the discrepancies in the evaluation represented a non-compliance with NFPA 805, but that functionality was assured based on existing instructions in the daily operator rounds procedure to check and, if necessary, manually rotate and backwash the strainers approximately once every 12 hours. Inspectors also reviewed plant operating histories and additional licensee evaluations which supported a conclusion that the loss of power to the strainers would likely not have an impact on safe shutdown equipment within the first 24 hours of a fire.

Analysis: The licensee's failure to perform an adequate engineering analysis for multiple fire areas to determine the effects of fire on the ability to achieve the nuclear safety performance criteria was a performance deficiency. This performance deficiency was determined to be more-than-minor because it was associated with the protection against external factors attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective in that failure to adequately analyze the effects of fire-damaged cables for the EECW strainers and backwash valves impacted the objective of ensuring the reliability of the EECW system during a fire. The finding was screened in accordance with NRC IMC 0609, "Significance Determination Process", Attachment 4, "Initial Characterization of Findings," dated October 7, 2016, which determined that an IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated September 20, 2013, review was required because it potentially affected the ability to reach and maintain safe and stable conditions in case of a fire. Using the Phase 1 Screening, the finding was assigned a category of Post-fire Safe Shutdown (SSD). The inspectors used step 1.4, task 1.4.5 "Post-fire Safe Shutdown" to determine the finding to be of very low safety significance (Green) because the finding did not affect the ability to reach and maintain a stable plant condition within the first 24 hours. The finding had a cross-cutting aspect of Avoid Complacency (H.12) within the cross-cutting area of Human Performance because the licensee did not recognize that historical assumptions about long-term strainer functionality could contain mistakes and latent issues during development of the NSCA.

Enforcement: Browns Ferry Nuclear Plant Units 1, 2 and 3 Renewed License Numbers DPR-33, 52, 68, conditions 2.C.(13), 2.C.(14), and 2.C.(7), required the licensee to implement and maintain in effect all provisions of the approved fire protection program that complied with 10 CFR 50.48(c) "National Fire Protection Association Standard NFPA 805". NFPA 805 Section 2.4.2.4 stated that an engineering analysis shall be performed for each fire area to determine the effects of fire on the ability to achieve the nuclear safety performance criteria. Contrary to the above, since February 2013, the licensee failed to perform an adequate engineering analysis for fires in multiple fire areas to determine the effects of fire on the ability to achieve the nuclear safety performance criteria. As an immediate corrective action, the licensee performed plant-specific analyses to determine the effects of fire on the functionality of EECW strainers and EECW system. The licensee also added recovery actions to the Fire Safe Shutdown procedures to ensure strainer functionality during fire scenarios. The licensee entered the violation into the licensee's corrective action program as CRs 1260785 and 1263434.

This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. (NCV 05000259/260/296/2017002-01, Inadequate Fire Risk Evaluation for Postulated Fires Affecting EECW Strainers)

1R06 Flood Protection Measures (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed internal flood protection measures for the Unit 1 and 2 Diesel Generator Building internal flood design to verify that flood mitigation plans were consistent with the design requirements and risk analysis assumptions and that equipment essential for reactor shutdown was properly protected from a flood caused by pipe breaks in the rooms/building. Specifically, the inspectors reviewed the licensee's moderate energy line break flooding study to understand the licensee's flood mitigation strategy, reviewed licensee drawings and then verified that the assumptions and results remained valid. The inspectors walked down the areas to verify the assumed flooding sources, adequacy of common area drainage, and flood detection instrumentation to ensure that a flooding event would not impact reactor shutdown capabilities. The inspectors completed one Internal Flooding sample as defined in Inspection Procedure 71111.06.

b. Findings

Introduction: An NRC-identified Green violation of 10 CFR 50, Appendix B, Criterion III was identified for the licensee's failure to verify the adequacy of the Unit 1 and 2 diesel building emergency drain pipe to mitigate a postulated internal flood.

Description: Inspectors evaluated the licensee's design review for the Unit 1 and 2 diesel building emergency drain pipe contained in calculation MDQ00004020110008 "Flow Capacity of the Diesel Generator Building Emergency Drain Piping". The 24 inch diameter drain pipe was designed to passively mitigate flooding from a critical crack of an Emergency Equipment Cooling Water pipe in the diesel building by providing a gravity drainage pathway to the outside of the building. The licensee concluded that the drain pipe had a flow capacity of 1274 gallons per minute with the building flooded to a height of 1 inch. This capacity was substantially greater than the postulated 500 gallons per minute leak rate from the EECW pipe. Inspectors compared the assumptions in the calculation with actual field measurements and discovered several discrepancies with the design review: 1) the capacity of the drain was calculated based on the full 415.5 square inch area of the 24-inch pipe; however, with only 1 inch of water in the building, just 2.2 square inches of the drain pipe area would be submerged, 2) water could not enter the drain until the water level accumulated above 1 inch because the drain pipe penetrated the wall at an elevation higher than depicted on plant drawings, 3) a grating was present inside the flow area of the drain pipe which would obstruct water flow into the drain, 4) the treatment of the drainage as flow through an orifice instead of open channel flow through a partially filled pipe was not justified, 5) a previous licensee evaluation for PER 10-268624 concluded the accumulation would be 4.75 inches; however, the methods and assumptions used in that evaluation were replaced by the non-conservative methods in the final design review. After considering the cumulative effects, inspectors determined that the water could accumulate to approximately

5.5 inches. By letter CNL-15-041, dated March, 21, 2015, (ML15072A130), the licensee previously informed the NRC that water accumulations of up to 6 inches could occur before safety-related equipment would be adversely affected. However, more recent evaluations have determined that accumulations of up to 8 inches would be acceptable. The limiting components were the diesel generator motor driven soakback and lube oil circulating pumps. Inspectors also identified that the output voltage cables from the diesel generator would be submerged by water accumulations to 5.5 inches; however, the licensee confirmed that the cables were not susceptible to damage from short-term water submergence. The emergency drain pipe remained adequately designed despite the non-conservative assumptions in the calculation because the diesel generators would remain protected to 8 inches.

Analysis: The inspectors determined that the failure to verify the adequacy of design of the Unit 1 and 2 diesel building emergency drain pipe was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective because non-conservative assumptions in calculation MDQ00004020110008 resulted in inaccurate conclusions about the capacity of the drain and the resulting water accumulation in the building. This finding was evaluated in accordance with IMC 0609, Att. 4, “Initial Characterization of Findings,” issued October 7, 2016, for Mitigating Systems, and IMC 0609, App. A, “The SDP for Findings At-Power,” issued June 19, 2012. The finding was determined to be Green because it represented a deficiency affecting the design of the drain piping, but it maintained its functionality because additional evaluation showed that the resulting water accumulation would not affect any safety related equipment. No cross-cutting aspect was assigned because it was not considered to be reflective of current licensee performance since the performance deficiency occurred more than three years ago.

Enforcement: 10 CFR 50, Appendix B, Criterion III required, in part, that design control measures shall provide for checking the adequacy of design, such as by the performance of design reviews. Contrary to the above, since May 6, 2011, the design review performed for the diesel generator building emergency drain piping did not verify adequacy of the design because of non-conservative assumptions. As an immediate corrective action, the licensee reevaluated the potential water accumulation and concluded the diesel generators were still protected. The licensee entered the violation into their corrective action program as CR 1303737. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. (NCV 05000259/260/2017002-02, Non-conservative Assumptions in Emergency Drain Capacity Design Review)

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the heat sink performance of heat exchangers that are required to remove decay heat and/or provide cooling water for risk-significant or safety-related equipment. This review focused on verifying that the licensee was adequately identifying and resolving heat sink performance problems that could result in initiating events or affect multiple heat exchangers in mitigating systems. Inspectors assessed whether the licensee had identified potential deficiencies which could mask degraded performance or result in common cause problems that have the potential to increase risk. The inspectors completed two heat sink performance inspection samples as defined in Inspection Procedure 71111.07.

- Unit 1 High Pressure Coolant Injection Lube Oil Heat Exchanger
- 3D Emergency Diesel Generator Jacket Water Coolers

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification and Performance (71111.11)

.1 Licensed Operator Requalification

a. Inspection Scope

On May 15, 2017, the inspectors observed a licensed operator training session for the Group 1 operating crew on the Unit 2 Simulator involving a recirculation pump trip, ATWS, and failed fuel. The inspectors evaluated the following attributes to assess the performance of the licensed operators':

- 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 normal and emergency procedures
- Timely control board operation and high-risk operator actions
- Timely oversight and direction provided by the shift supervisor, including implementing appropriate technical specifications and emergency plan notifications
- Group dynamics involved in crew performance

The inspectors reviewed the in-process critiques performed by the licensee evaluators, and verified that licensee-identified issues were comparable to issues identified by the inspector. The inspectors reviewed simulator physical fidelity. This activity constituted one Observation of Requalification Activity inspection sample as defined in Inspection Procedure 71111.11.

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 covering Conduct of Operations, Plant Operations and Power Maneuvering.

Inspectors utilized activities such as post-maintenance testing, surveillance testing and other 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 as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the specific structures, systems and components (SSC) within the scope of the Maintenance Rule (MR (10CFR50.65) with regard to some or all of the following attributes, as applicable:

(1) Appropriate work practices
(2) Identifying and addressing common cause failures
(3) Scoping
(4) Characterizing reliability issues
(5) Tracking unavailability
(6) Balancing reliability and unavailability
(7) Trending key parameters for condition monitoring
(8) System classification and reclassification
(9) Appropriateness of performance criteria
(10) Appropriateness and adequacy of 50.65 (a) (1) goals, monitoring and corrective actions
(11) Quality control aspects

The inspectors compared the licensee's performance against site procedures. The inspectors reviewed, as applicable, work orders, surveillance records, CRs, 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. Documents reviewed are listed in the attachment. This activity constituted three Maintenance Effectiveness inspection samples as defined in Inspection Procedure 71111.12: Two routine maintenance effectiveness samples and one quality control sample.

- 1C RHR room cooler failure to start

- U2 RCIC out of service due to broken packing gland on vacuum pump
- Quality control during 3ED diesel generator maintenance window

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

For planned online work and/or emergent work that affected the combinations of risk significant systems listed below, the inspectors examined 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. 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 two Maintenance Risk Assessment inspection samples as defined in Inspection Procedure 71111.13.

- Emergent B Residual Heat Removal Service Water (RHRSW) header isolation to repair instrumentation line through-wall leak
- Planned risk associated with the Main Bank Battery-1 unavailability during an extended discharge test. The Equipment out of Service (EOOS) multiplier was at the yellow risk threshold.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessment (71111.15)

a. Inspection Scope

The inspectors reviewed the operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedures to ensure that the licensee's evaluation met procedure requirements. 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 reviewed CRs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. This activity constituted seven Operability Evaluation inspection samples as defined in Inspection Procedure 71111.15.

- Unit 1 HPCI cooling water relief 1-73-0574 found lifting (CR 1288289)
- Unit 1 and 2 C EDG high vibrations (CR 1265908)

- Unit Common concern for tornado generated missile hazard possibly affecting Unit 2 Main Steam Isolation Valves (MSIVs), Unit 1 and 3 Main Steam Relief Valves and various electrical boards (CR 1306987)
- Unit 3 3EB EDG redundant start circuit timing relay not within required range (CR 1281384)
- Unit 2 and 3 RCIC overspeed tappet nut engagement (CR 1289509)
- Diesel Generator 'C' degraded voltage signal timer missing seismic restraining strap (CR 1244680)
- Unit 1 control rod 34-35 declared slow due to high temperature with two adjacent control rods (34-31 and 38-39) previously declared slow (CR 1311713)

b. Findings

No findings were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors witnessed and reviewed post-maintenance tests (PMTs) listed below to verify that procedures and test activities confirmed Structure, System, or Component (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 functions 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. The inspectors witnessed and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety functions. The inspectors verified that problems associated with PMTs were identified and entered into the Corrective Action Program (CAP). This activity constituted two PMT inspection samples as defined in Inspection Procedure 71111.19.

- Unit 3 – Emergent weld repair to 3C EDG oil cooler (leak) and other 4-year planned maintenance items, WO 118802981
- Unit 2 – RHRSW Heat Exchanger B Discharge Valve testing following a RHR Loop II maintenance outage, WO 118063208

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed portions of, and/or reviewed completed test data, for the following surveillance tests of risk-significant and/or safety-related systems to verify that the tests met technical specification 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 four Surveillance Testing inspection samples: two routine tests, and two in-service tests as defined in Inspection Procedure 71111.22.

Routine Surveillance Tests:

- 2-SR-3.5.1.7, Unit 2 HPCI Main and Booster Pump Flow Rate Test
- 0-SR-3.7.3.4, Control Bay Habitability Zone Pressurization Test

In-service Tests:

- 0-SI-4.5.C.1(B3-COMP) RHRSW Pump B3 IST Comprehensive Pump Test
- 0-SI-4.5.C.1(D SMP) RHRSW Room D Sump Pump Test

b. Findings

Introduction: An NRC-identified Green NCV of 10 CFR Part 50, Appendix B, Criterion III was identified for the licensee's failure to correctly translate the design basis of the EECW system into technical instruction 0-TI-579(EECW).

Description: On April 12, 2017, IST evaluation 17-0-IST-023-654 was performed after an annual comprehensive test revealed the B3 EECW pump was in the Low Alert Range with only 17 GPM margin to the low required action range. The pump was re-baselined in accordance with technical instruction 0-TI-579(EECW), "EECW Pump Baseline Data Acquisition and Evaluation." When re-baselining a pump per ISTB-6200 of the ASME-OM code (2004 edition), a system level evaluation is required in order to verify operational readiness. The system level evaluation incorporated into TI-579 (EECW) included a note to not allow the Low Alert Range limit to drop below 2400 GPM because that flowrate represented the system's design basis minimum pump curve value at the differential head used in the test. The inspectors identified that this value did not account for instrument uncertainties. NPG-SPP-09.1.21, "Inservice Testing Program Evaluations and Reference Values," required instrument uncertainties to be included when tests were performed within +/-20% of the system's design/analysis limit. The inspectors identified that other effects, such as operating the pump motor at the reduced diesel generator frequency, also were not included. Operation at 59Hz was allowed by operating procedures 0-OI-82, "Standby Diesel Generator System" and 0-AOI-57-1A "Loss of Offsite Power (161 and 500 KV) / Station Blackout." The effect of frequency variation was previously evaluated by the licensee in design review calculation NDQ099920100006, "Diesel Generator Frequency Variation Evaluation," which determined there would be an acceptable reduction in flow; however, this flow reduction was not integrated into the minimum acceptable pump curve or the system level evaluation.

When the pump's performance was adjusted for the cumulative effects, portions of the pump curve for the B3 EECW pump were less than the minimum allowed. According to drawing 1-47E858-1-ISI, pumps operating below the minimum curve should be declared inoperable. A subsequent engineering evaluation showed that although the pump would be below the minimum curve at the specified differential head used in performing the test, data recently acquired from 0-TI-345(EECW), "EECW Pump Curve Data Acquisition," showed that the pump would actually perform with some positive margin at the lower differential head that is expected during a design basis event. The reduction in available margin would be approximately 30 percent based on current performance. If the pump were allowed to degrade to the minimum allowed by 0-TI-579(EECW), the available margin would be reduced by approximately 60 percent.

Analysis: The failure to correctly translate the EECW system design bases into technical instruction 0-TI-579(EECW) was a performance deficiency. Specifically, the effects of instrument uncertainty and diesel frequency variations were not considered when establishing the minimum allowed inservice test low alert limits. The performance deficiency was more-than-minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective in that, when the effects of instrument uncertainty and diesel frequency variations were considered, there was a reasonable doubt on the operability of the B3 EECW pump since portions of the adjusted pump curve would be below the minimum pump curve established in the design basis calculation. Additionally, there was a significant reduction in available margin for the pump under design basis conditions. The inspectors evaluated the finding with IMC 0609, Att. 4, “Initial Characterization of Findings,” issued October 7, 2016, for Mitigating Systems, and IMC 0609, App. A, “The SDP for Findings At-Power,” issued June 19, 2012. The inspectors determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design of a mitigating system, but the pump maintained its operability based on the additional engineering evaluations which demonstrated satisfactory system performance at the reduced pump performance level. The inspectors determined that the finding had a cross-cutting aspect of Human Performance (H.6) within the cross-cutting area of Design Margins because engineers did not demonstrate the characteristic of ensuring that design margins were guarded and changed only through a systematic and rigorous process.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” stated, in part, that “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into drawings, procedures and instructions.” Contrary to the above, since March 2012, the licensee’s design control measures failed to correctly translate the design basis of the EECW system into technical instruction 0-TI-579(EECW), which was used to establish the low alert limits for inservice testing. As an immediate corrective action, the licensee evaluated the operability of the pump and initiated corrective action to make changes to the test criteria and/or the system design analysis. The licensee entered the violation into the licensee's corrective action program as CR 1288208. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. (NCV 05000259/260/296/2017002-03, Failure to Assure EECW Design Basis Capability)

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed Emergency Planning (EP) Radiological Emergency Plan (REP) training drills that contributed to the licensee’s Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures. This drill was intended to identify any licensee weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed two emergency response operations, one in the Simulated Control Room on May 3, 2017 and one in the Technical Support Center (TSC) on May 24, 2017, to verify event classification and notifications were done in accordance with the licensee's procedures. The inspectors attended the post-drill critiques to compare any inspector-observed weaknesses with those identified by the licensee in order to verify whether the licensee was properly identifying EP related issues and entering them into the CAP. This constituted two samples, one Simulator based and one TSC based sample as defined in Inspection Procedure 71114.06.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones; Initiating Events, Mitigating Systems

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Mitigating Systems

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following PIs. The inspectors examined the licensee's PI data for the specific PIs listed below for the second quarter of 2016 through the first quarter of 2017. The inspectors reviewed the licensee's data and graphical representations as reported to the NRC to verify that the data was correctly reported. The inspectors validated this data against relevant licensee records (e.g., CRs, Daily Operator Logs, Plan of the Day, Licensee Event Reports, etc.), and assessed any reported problems regarding implementation of the PI program. The inspectors verified that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors used the Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to ensure that industry reporting guidelines were appropriately applied. This activity constituted nine PI inspection samples, as defined in Inspection Procedure 71151.

- Units 1, 2, and 3 Safety System Functional Failures (SSFF)
- Units 1, 2, and 3 HPCI Mitigating System Performance Indicator (MSPI)
- Units 1, 2, and 3 RCIC MSPI

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution of Problems (71152)

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

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 CR reports, and periodically attending Management Review Committee and Plant Screening Committee meetings.

.2 Annual Follow-up of Selected Issues: Extent of Condition Review for CR 692133

a. Inspection Scope

For this Annual Follow-up the inspectors assessed licensee performance against selected attributes listed in section 03.06 of Inspection Procedure 71152 to specifically ensure that corrective actions were appropriately focused to address the root and contributing causes for a SCAQ. Inspectors reviewed the licensee's corrective actions for CR 692133 related to the October 2012 wedge pin failure of the 1FCV-73-2 HPCI containment isolation valve. Associated with this review was a review of the site's root cause efforts, the Flowserve Part 21 # 48797 documentation and a recent similar failure at LaSalle Station. This inspection constituted one focused Annual Follow-up of Selected Issues sample as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

.3 Semi-annual Trend Review:

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP and other associated programs and documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also included licensee trending efforts and licensee human performance results. The inspectors' review nominally considered the six-month period of January through June 2017. The inspectors reviewed licensee trend reports and other maintenance and health reports, in order to determine the existence of any adverse trends that the licensee may not have previously identified. This inspection constituted one Semi-annual Trend Review inspection sample as defined in IP 71152.

b. Observations and Findings

No violations were identified. The licensee had identified trends and appropriately addressed them in their CAP. The inspectors observed that the licensee had performed a detailed review. The licensee routinely reviewed cause codes, involved organizations, established key words and system links to identify potential trends in their data. The inspectors compared the licensee process results with the results of the inspectors' review. Trends that have been identified by the inspectors and reported to the licensee were appropriately entered into the licensee's trending program. Noteworthy licensee-identified trends included:

- All three Units' Reactor Water Cleanup System pumps continue to have seal failures. About one failure every month (CR 1310246)
- A declining trend in Human Performance for both the Operations and Security departments (CRs128814,1297466,1297267, and 1268411)
- NRC related noncited violations related to valve maintenance issues CR 1288502

Noteworthy NRC-identified adverse trends included:

- Monitoring Programs such as Flood Prevention, Raw Water piping systems integrity monitoring (corrosion issues), and MOV testing early identification actions not being fully effective. This conclusion is supported by QA, other non-line groups, and self-revealing events identifying undetected problems / deficiencies.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report (LER) 05000260/2017-003-00, Manual Reactor Scram During Startup Due to Multiple Control Rod Insertion

a. Inspection Scope

This LER was associated with the Unit 2 manual reactor scram that occurred during a reactor startup on March 29, 2017. The inspectors reviewed the root cause report and discussed the issue with appropriate members of plant staff. The cause of the scram was attributed to an Intermediate Range Monitor (IRM) 'G' faulty pre-amp combined with IRM signal spikes associated with manipulation of the scram reset switch that affected the IRM 'F' instrument. This condition was documented in the licensee's corrective action program as CR 1278595. The inspectors reviewed the CR and the licensee's corrective actions.

b. Findings

Introduction. The inspectors reviewed a Green self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion XVI, 'Corrective Action,' for the licensee's failure to establish measures to assure that corrective action was taken to preclude repetition of a SCAQ. The licensee failed to implement corrective actions associated with the scram reset switch and other ground induced electronic noise sources.

Description. On May 24, 2012, during a Unit 3 startup, a control room operator inadvertently ranged an IRM down instead of up resulting in a half-scram in the Reactor Protection System (RPS) 'B' trip channel. Subsequently, the IRM was properly ranged and plant operators reset the half-scram. Coincident with resetting the half-scram, an electrical spike was received on IRM 'A' in the RPS 'A' trip channel resulting in rod insertion for the Group 1 and 4 control rods. The licensee determined the direct cause of the Unit 3 event was electronic noise on the control room common ground which, through a degraded connector, caused an electronic spike on the 3A IRM. The licensee classified the condition a SCAQ requiring a root cause evaluation and corrective actions to prevent repetition per Procedure NPG-SPP-03.1, "Corrective Action Program." An extent of condition review was performed to cover IRMs in the other operating units. It was then identified that IRMs and Source Range Monitors (SRMs) were vulnerable to noise from the scram reset switch when resetting a half-scram. One of the corrective actions to prevent recurrence was to identify the source of the electronic noise so noise suppression could be performed. However, testing to identify the source caused half-scrams to occur during power operation and was stopped. No further testing was conducted and the licensee's noise suppression efforts were limited to SRMs only. IRM spiking continued to occur on all three units indicating that electronic noise on the control room common ground was still present.

On March 29, 2017, during a Unit 2 startup, IRM 'G' drifted low. When control room operators ranged IRM 'G' down one position, the IRM spiked upscale causing a reactor half-scram on RPS 'A'. After verifying the IRM High-High trip cleared, the operators attempted to reset the half-scram. When the scram reset switch was turned, a trip signal from IRM 'F' was generated on RPS 'B' resulting in a Groups 1 and 4 only control rod insertion. Due to only half the control rods inserting, control room operators inserted a manual reactor scram in accordance with AOI-100-1. The licensee determined the scram was due to electronic noise on the control room common ground which was not corrected from the May 24, 2012, scram. Based on this determination, the licensee classified this event as preventable and classified it as a repeat event.

Analysis. The failure to ensure corrective actions were taken for a SCAQ to preclude repetition of a reactor scram associated with electronic noise was a performance deficiency. This performance deficiency is more-than-minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective in that the licensee failed to implement corrective actions to address electronic noise after the May 24, 2012, reactor scram. Using Inspection Manual Chapter 0609 Appendix A, the inspectors determined that this finding is of very low safety significance (Green) because it did not involve the loss of mitigation equipment per Exhibit 1.B "Transient Initiators." The inspectors determined that the finding had a cross-cutting aspect of Challenge the Unknown (H.11) within the cross-cutting area of Human Performance because the licensee failed to resolve the unknown noise paths to ensure that scram vulnerabilities were corrected.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, 'Corrective Action,' requires, in part, that for SCAQs, the licensee establish measures to assure that corrective action is taken to preclude repetition. Contrary to the above, from May 24, 2012, to March 29, 2017, the licensee failed to establish measures to ensure that corrective action was taken for a SCAQ to preclude repetition. The licensee failed to implement corrective actions to address a SCAQ and preclude repetition of a plant scram due to electronic noise. The licensee entered this issue into their corrective action program as CR 1278595. Because the finding is of very low safety significance (Green) and has been entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000260/2017002-04, 'Failure to Implement Corrective Actions to Prevent the Recurrence of a Reactor Scram Due to IRM spiking.'

.2 (Closed) LER 05000259/2017-001-00, Signal Timer for 4kV Shutdown Board C Inoperable for Longer Than Allowed by Technical Specifications due to Detached Restraining Strap

On December 21, 2016, personnel discovered a detached seismic restraining strap on the 4kV shutdown board 'C' degraded voltage relay timer during a surveillance. A past operability evaluation determined the timer was inoperable from October 5, 2016 until December 22, 2016 which exceeded the Technical Specification allowed outage time.

The cause of the event was most likely the result of human error during installation of the relay mounting screws and failure to identify the inadequate installation during subsequent tests and inspections. The inspectors reviewed the LER and the licensee's corrective actions. The enforcement aspects of this violation are discussed in Section 4OA7.

.3 (Closed) LER 05000260/2016-001-01, High Pressure Coolant Injection Safety System Functional Failure due to a Blown Fuse and a Failed Relay

The report was reviewed based on the changes that were made to the initial LER. Review of the previous report was documented in Browns Ferry inspection report 05000259, 260, 296/2016003 (ML16315A108). The changes to the report included additional failure analysis results, corrective actions and an enhanced timeline of events. The inspectors reviewed the LER revision and did not identify any additional findings.

.4 (Closed) LER 05000260/2017-001-00, High Pressure Coolant Injection System Safety Functional Failure due to a Blown Fuse

On February 16, 2017, a spurious failure of a fuse protecting certain HPCI system control circuits occurred which rendered the system inoperable. Within one hour, the failed fuse was identified and replaced. After monitoring the circuit for satisfactory currents, the system was declared operable. The system was inoperable for approximately 31 hours. The cause of the event was most likely the result of a spontaneous failure due to solder creep affecting the connection between the fuse's internal resistor and its tension/retraction spring. The inspectors reviewed the LER and the licensee's corrective actions.

.5 (Closed) LER 05000259/260/296-2016-003-00, Fire Safe Shutdown (SSD) Procedures Do Not Consider Potential for Fire-Induced Failure of 4kV Shutdown Board Under-Voltage Trip Functions

On August 03, 2016, during circuit analysis review of the SSD strategy for FA 9, it was determined that the SSD analysis did not consider the potential for fire-induced failure of the credited 4kV Shutdown Board undervoltage trip function for emergency diesel generator (EDG) power supply alignments. When relying on an EDG for SSD, the licensee's SSD strategy credited load shedding of the associated shutdown board, such that only essential loads will be loaded to the EDG. This is to prevent overloading the EDG. Fire-induced failure of the board under-voltage 27S relay function could result in the credited shutdown board's loads not shedding upon a shutdown board undervoltage condition. If the nonessential loads are not automatically removed from the credited shutdown board, operators would need to perform additional actions to trip board loads. These actions were incorporated into the Fire Safe Shutdown procedures as part of corrective actions for NRC violation NCV 05000259, 260, 296/2016011-02, Failure to Adequately Identify and Evaluate All Circuit Failures for NSCA Credited Equipment. The inspectors reviewed the LER and the licensee's corrective actions.

4OA5 Other Activities

.1 Operation of an Independent Spent Fuel Storage Installation (ISFSI) at Operating Plants (IP 60855.1)

a. Inspection Scope

The inspectors reviewed changes made to the ISFSI programs and procedures, including those associated with 10 CFR 72.48, "Changes, Tests, and Experiments," screens and evaluations to verify that changes made were consistent with the license and/or certificate of compliance. The inspectors reviewed records to verify that the licensee recorded and maintained the location of each fuel assembly placed in the ISFSI. This activity constituted one semi-annual Operation of an ISFSI inspection sample, as defined in IP 60855.1.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On July 14, 2017, the resident inspectors presented the quarterly inspection results to Mr. Steve Bono, Site Vice President and other members of the licensee's staff, who acknowledged the findings. The inspectors confirmed that proprietary information was controlled to protect it from public disclosure.

4OA7 Licensee-Identified Violations

The following licensee-identified violation of NRC requirements was determined to be of very low safety significance and met the NRC Enforcement Policy criteria for being dispositioned as a non-cited violation.

- 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, between October 5, 2016, and December 22, 2016, the 4kV shutdown board 'C' degraded voltage relay timer was not installed in accordance with MAI-3.8, "Installation of Electrical Components." The failure to install mounting screws of an appropriate length with suitable thread engagement for the seismic restraining strap resulted in the relay being inoperable for longer than the Technical Specification allowed outage time. The licensee entered the violation into the corrective action program as CR 1244680 and replaced the damaged mounting screw and installed the seismic restraining strap. Using an exposure time of 78 days, the change in core damage frequency was conservatively estimated to be less than 4E-8 per year. The most dominant core damage sequences were those involving the loss of the high pressure injection systems. The significance of the finding was limited because it did not affect the ability of the diesel generator to automatically start under loss of offsite power conditions and it did not affect the ability of operators to manually start the diesel generator in response to degraded voltage conditions. The inspectors determined the finding was Green.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Bono, Site Vice President
L. Hughes, General Manager, Site Operations
J. Paul, Acting Director of Safety and Licensing
M. Oliver, Acting Nuclear Site Licensing Manager
M. McAndrew, Manager of Operations
B. Bruce, Work Management Director
L. Slizewski, Superintendent of Operations
M. Kirschenheiter, Assistant Director for Site Engineering
D. Drummonds, Program Engineer
J. Barker, Operations Superintendent
J. Smith, System Engineer
T. Stafford, NFPA-805 Program Manager
R. Guthrie, Emergency Diesel System Engineer
T. Womack, TVA Corporate Electrical Design Program Manager
A. Taylor, TVA Corporate Design Engineering Senior Manager
R. Cox, TVA Corporate Electrical Design Engineering Manager
R. Beck, Engineering FIN team Manager
J. Addison, TVA EP Drill Coordinator
B. Tidwell, EP Manager

LIST OF REPORT ITEMS

Opened and Closed	
NCV 05000259/260/296/2017002-01	Inadequate Fire Risk Evaluation for Postulated Fires Affecting EECW Strainers (Section 1R05)
NCV 05000259/260/2017002-02	Non-conservative Assumptions in Emergency Drain Capacity Design Review (Section 1R06)
NCV 05000259/260/296/2017002-03	Failure to Assure EECW Design Basis Capability (Section 1R22)
NCV 05000260/2017002-04	Failure to Implement Corrective Actions to Prevent the Recurrence of a Reactor Scram Due to IRM spiking (Section 4OA3.1)
Closed	
LER 05000260/2017-003-00	Manual Reactor Scram During Startup Due to Multiple Control Rod Insertion (Section 4OA3.1)
LER 05000259/2017-001-00	Signal Timer for 4kV Shutdown Board C Inoperable for Longer Than Allowed by Technical Specifications due to Detached Restraining Strap (Section 4OA3.2)
LER 05000259/2016-001-01	High Pressure Coolant Injection Safety System Functional Failure due to a Blown Fuse and a Failed Relay (Section 4OA3.3)
LER 05000260/2017-001-00	High Pressure Coolant Injection System Safety Functional Failure due to a Blown Fuse (Section 4OA3.4)
LER 05000259/260/296/2016-003-00	Fire Safe Shutdown Procedures Do Not Consider Potential for Fire-Induced Failure of 4kV Shutdown Board Under-Voltage Trip Functions (Section 4OA3.5)

LIST OF DOCUMENTS REVIEWED

Section 1R04

Procedures

2-OI-73, High Pressure Coolant Injection System, Rev. 97
3-SR-3.1.7.7, SLC Functional Test, (Rev 34)
3-SR-3.3.6.1 (GRP 2, 3 & 8) RWCU interlock with SLC

Drawings

0-17W300-9, Mechanical Isometric Emer. Equip. Cooling Water, Rev. 1
0-17W405-3, Mechanical Building Drainage, Rev. 3
0-47E851-1, Flow Diagram Drainage, Rev. 33
0-47E851-4, Flow Diagram Drainage, Rev. 18
0-47E851-6, Flow Diagram Drainage, Rev. 2
1-47E859-1, Flow Diagram Emergency Equipment Cooling Water, Rev. 97
1-47E858-1, Flow Diagram RHR Service Water System, Rev. 72
2-47E812-1, Flow Diagram High Pressure Coolant Injection, Rev. 73
2-47E859-1, Flow Diagram Emergency Equipment Cooling Water, Rev. 36
3-45E779-3, Wiring Diagram 480v Shutdown Aux Power Schematic Diagram (SLC)
3-47E610-63-1, U3 SLC heaters
3-47E859-1, Flow Diagram Emergency Equipment Cooling Water, Rev. 41
3-47E859-2, Flow Diagram Emergency Equipment Cooling Water, Rev. 28

Other Documents

BFN-50-7067, Emergency Equipment Cooling Water System, Rev. 22
MDQ0000672013000125, Evaluation of EECW Component Flow Rates with the Most Limiting Pump Configuration, Rev. 0
MDQ0067880346, EECW, Headloss, Pressure, Rev. 5
MDQ0067930028, EECW System Pressure Drop – Multiflow, Rev. 8
CD-Q2063-894704, Evaluation of Nozzle Loads on SLC test tank, and storage tank

Section 1R05

Procedures

0-FSS-16-1, Control Building EL 593’, 606’, 617’ and 635’, Rev. 6
0-FSS-20, U-1 & 2 Diesel Generator Building EL 565’ – 583’, Rev. 3
0-GOI-300-1/ATT-12, Outside Operator Rounds Log, Rev. 237
0-OI-67, Emergency Equipment Cooling Water System, Rev. 114
0-SIMI-67B, Emergency Equipment Cooling Water System Scaling and Setpoint Documents, Rev. 41
1-ARP-9-20A, Panel 1-9-20-XA-55-20A, Rev. 31
FPR-VOLUME 2, Fire Protection Report Volume 2, Rev. 55

Drawings

0-45E614-7, Wiring Diagram 120V AC/250 DC Valves & Misc Schematic Diagram, Rev. 34
0-45E771-5, Wiring Diagram 480V Diesel Aux Power Schematic Diagram, Rev. 28
0-47E216-106, Ignition Source Drawings Plan El. 606.0 & 621.25, Rev. 0
0-47E610-67-2, Mechanical Control Diagram Emergency Equipment Cooling Water System, Rev. 38
1-47E610-67-1, Mechanical Control Diagram Emergency Equipment Cooling Water System, Rev. 44
2-47W845-1, Mechanical Flow Diagram – Essential Raw Cooling Water System, Rev. 54

Other Documents

Apparent Cause Evaluation Report for PER 381569, 3D Diesel Generator Inoperable due to low EECW flow
BFN-VTD-K143-0020, Installation and Maintenance Instructions for Kinney Automatic Self-Cleaning Strainers Model AV Series, Rev. 7
BFN-VTM-K143-0010, Vendor Technical Manual for Kinney Automatic Self Cleaning Strainers Model AV Series, Rev. 9
BFN-VTM-K143-0050, Instruction Manual for S. P. Kinney Model AV Series Motorized Automatic Self-Cleaning Strainers Class 1, Rev. 0
CDEs 1063, 1064, 1596
CRs 1260785, 1263434, 1312757
EDQ099920110010, NFPA 805 – Nuclear Safety Capability Analysis, Rev. 33
Functional Evaluation for CR 1260785
Functional Evaluation for CR 1263434
Jensen Hughes Report NWL06045-RPT-001, BFN EECW Strainer Performance During Loss of Automatic Strainer Operation, Rev. 0
Jensen Hughes Report NWL06045-RPT-002, BFN EECW Strainer Evaluation, Rev. 0
LER 05000321/1980-103
LER 05000368/1991-012-00
MDN0009992012000100, Browns Ferry Nuclear Power Plant, Units 1, 2, and 3, Fire Risk Evaluations, Rev. 6
MDN0009992012000100, Browns Ferry Nuclear Power Plant, Units 1, 2, and 3, Fire Risk Evaluations, Rev. 0
Memorandum from L. W. Boyd, Browns Ferry Nuclear Plant (BFN) – Emergency Equipment Cooling Water (EECW) Strainers – Appendix R, dated December 16, 1988, RIMS B44881216006
NDN00006720070013, SY.06 – BFN Probalistic Risk Assessment – Emergency Equipment Cooling Water System, Rev. 51
NFPA 805 Fire Protection Requirements Manual, Rev. 2
NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition
NUREG-1461Regulator Analysis for the Resolution of Generic Issue 153: Loss of Essential Service Water in LWRs
Operator Work Around 0-067-OWA-2011-0056
PERs 1020900, 157394, 365232
Raw Water Inspection 110960920
SQN-67-D053, HCG-JDH-020188, ERCW Strainer Clogging Rate Analysis, Rev. 0 RIMS B44880216001

Section 1R06

BFN-50-720, Evaluating the Effects of a Pipe Failure Outside Containment, Rev. 1
BFN-50-C-7105, Pipe Rupture, Internal Missiles, Internal Flooding, and Vibration Qualification of Piping, Rev. 12
CR 1303737
DED-TM-PF1, Concluding Report on the Effects of Postulated Pipe Failure Outside of Containment for Unit 1 of the Browns Ferry Nuclear Plant, dated October 15, 1973
FSAR Section 10.16.4.6, Evaluation for Flooding due to Failure of Low Energy Piping Systems Outside Primary Containment
MDQ00004020110008, Flow Capacity of the Diesel Generator Building Emergency Drain Piping, Rev. 0

NDN00099920070031, IF – BFN Probabilistic Risk Assessment – Internal Flooding Analysis, Rev. 0
NDQ0999910033, Safe Shutdown Analysis, Rev. 27
Functionality Evaluation for PER 10-268624
Letter CNL-15-041, "Flood Hazard Reevaluation Report for Browns Ferry Nuclear Plant, Response to NRC 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, 2015, (ML15072A130)

Section 1R07
Drawings
0-105E2694, Process Diagram High Pressure Coolant Injection System for 3458 MWt, Rev. 1
1-47E812-1, Flow Diagram High Pressure Coolant Injection System, Rev. 43
1-47E812-2, Flow Diagram HPCI Oil System, Rev. 5

Procedures
3-SI-3.2.4(DG D) EECW Check Valve Test on Diesel Generator D, Rev. 12

Other Documents
BFN-VTD-L170-0010, Engineering Operating and Maintenance Data for Leslie Regulators and Controllers, Rev. 0
CR 1145553, 1288289, 1293692, 1294718
HPCI Lube Oil Cooler Outlet Temperature trends 2016-2017
HPCI Thrust Bearing Temperature trends 2016-2017
MDQ0000732012000040, HPCI Turbine Lube Oil Cooler Tube Plugging Evaluation, Rev. 0
MDQ0082000016, Diesel Generator Jacket Water Cooler Capacity and Tube Plugging, Rev. 2
PDO for CR 1288289
System Monitoring data trends for Diesel Generator EECW flows 2015-2017
WOs 115052074, 116152429, 116342872, 118020064

Section 1R11
Procedures
OPDP-1, Conduct of Operations, Rev. 38
OPDP-14, Operator Fundamentals, Rev. 1

Section 1R12
Procedures
0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting – 10CFR50.65, Rev. 50
MPI-0-082-INS003, Standby Diesel Engine 48 Month Inspection, Rev 0065
NPG-SPP-06.5, Foreign Material Control

Other Documents
CRs 1219045,1266004, 1283660, 1291943, 1292321
Past Operability Evaluation for CR 1291943
PDO for CR 1283660
System Health Report for system 074 for 1st quarter of 2017
System Health Report for system 074 for 1st quarter of 2017
WOs 117821215, 118130731. 118212303, 118557484, 118661080, 118717547

Section 1R13

Procedures:
0-OI-25, Raw Service Water System, Rev. 54
0-OI-26, High Pressure Fire Protection System, Rev. 101
1-ARP-9-3E, Annunciator Response Procedure for Panel 9-3, XA-55-3E, Rev. 32
BFN-ODM-4.18 Protected Equipment, Rev. 17
NPG-SPP-09.11.1 Equipment Out of Service Management, Rev. 12

Drawings:
1-47E858-1, Flow Diagram RHR Service Water System, Rev. 72
1-47E850-1, Flow Diagram Fire Protection and Raw Service Water, Rev. 25
1-47E850-2, Flow Diagram Fire Protection and Raw Service Water, Rev. 31
2-47E858-1, Flow Diagram RHR Service Water System, Rev. 28
2-47E850-1, Flow Diagram Fire Protection and Raw Service Water, Rev. 24
2-47E850-2, Flow Diagram Fire Protection and Raw Service Water, Rev. 27
3-47E850-1, Flow Diagram Fire Protection and Raw Service Water, Rev. 25

Other Documents:
CR 1292238
TO 0-2017-0001 section 0-0023-0025
Operator logs from May 4, 2017 through May 5, 2017
Protected equipment list May 05, 2017

Section 1R15

Procedures:
EPI-3-082-DGZ002, Diesel Generator 3B Redundant Start Test, Rev. 29
MSI-0-071-GOV001, Reactor Core Isolation Cooling (RCIC) Overspeed Trip Test, Rev. 33

Drawings:
3-45E767-5 Wiring Diagram Diesel Generators Schematic Diagram, Rev. 33
3-45E767-5-1 Wiring Diagram Diesel Generators Schematic Diagram, Rev. 4
3-45E768-3 Emergency Equipment Schematic Diagram Diesel Generator 3B, Rev. 15
3-C196F11012 Physical Schematic and Field Connections, Rev. 10

Other Documents:
BFN-VTD-E147-0010, Standby Diesel Generator Electrical Equipment Class 1, Rev. 16
Causal Analysis for CR 1265908
CRs 1265908, 1289509, 1311713
DCN 69168, Simplify Diesel Generator start logic; eliminate existing redundant start logic scheme, Rev. A
EDQ0082890129, Emergency Generator Essential Timing Relays Setpoint and Scaling Calculation, Rev. 1
OPL171.040, Reactor Core Isolation Cooling (RCIC) System, Rev. 27
Past Operability Evaluation (POE) for CR 1265908
PDO for CR 1289509
PER 102012
PER 119057
WO 118258480

Section 1R19

Procedures

2-SI-4.5.C.1(B), RHRSW Hx B Valves Quarterly IST Test, Rev. 8
3-SR-3.8.1.1(3C), rev 55
MPI-0-000-ACT001, Preventive Maintenance for Limitorque Operators, Rev. 54

Other Documents

WOs 118130668, 118132680, 118389111, 118802913

Section 1R22

Procedures

0-SI-4.5.C.1 (B3-COMP), RHRSW Pump B3 IST Comprehensive Pump Test, Rev. 10
0-SI-4.5.C.1 (D SMP), RHRSW Room D Sump Pump Test, Rev. 4
0-TI-345(EECW), EECW Pump Curve Data Acquisition, Rev. 4
0-TI-383, Evaluation of Test Results for the ASME OM Code Inservice Testing Program, Rev. 3
0-TI-579(EECW), EECW Pump Baseline Data Acquisition and Evaluation, Rev. 5
NPG-SPP-09.1.21, Inservice Testing Program Evaluations and Reference Values, Rev. 0-SR-3.7.3.4, Control Bay Habitability Zone Pressurization Test, Rev. 16
2-SR-3.5.1.7, Unit 2 HPCI Main and Booster Pump Set Developed Head and Flow Rate Test

Drawings

0-47E200-12-1, Control Bay Habitability Zone (CBHZ) Boundary, Rev. 2
1-47E858-1-ISI, ASME Section XI RHR Service Water System Code Class Boundaries, Rev. 30

Other Documents

BFN-50-7030A, Control Bay and Reactor Building Board Rooms Environmental Control Systems, Rev. 14
CRs 1283523, 1283843
IST Evaluation 17-0-IST-023-654
MDQ0000672013000125, Evaluation of EECW Component Flow Rates with the Most Limiting Pump Configuration, Rev. 2
MDQ002320100019, RHRSW System Hydraulic Analysis for Units 1, 2, & 3 RHR Heat Exchangers, Rev. 5
MDQ003020040025, Control Bay Habitability Zone Seismic Class II Boundaries and Max Allowable Breach Analysis, Rev. 3
MDQ0067910008, Flow Requirements of EECW-Fed Components, Rev. 18
MDQ0067930028, EECW System Pressure Drop – Multiflow, Rev. 8
WOs 117445289, 117446734, 118447203

Section 1EP6

Procedures

EPIP-1, Emergency Classification Procedure, Rev. 54
REP-Appendix A, Radiological Emergency Plan Appendix A, Rev. 108

Other Documents

CR 1300370, 1300372, 1300373, 1300374, 1300377, 1300390
Drill Report for the Browns Ferry Training Drill conducted on May 3, 2017
Drill Report for the Browns Ferry Training Drill conducted on May 24, 2017

Section 4OA1
CR 1263268
EACE for CR 1263268
EN 52557

Section 4OA2
Other Documents
Licensee generated Trend Report for January 01, 2017 to June 2017

Section 4OA3
Procedures
2-ARP-9-3F, Panel 9-3 2XA-55-3F, Rev. 34
MAI-3.8, Installation of Electrical Components, Rev. 10

Drawings
0-45E765-8, Wiring Diagram 4160V Shutdown Aux Power Schematic Diagrams, Rev. 25
0-731E761-5, Elementary Diagram Emergency Equipment, Rev. 18
2-45W2686-5, Wiring Diagrams, ECCS DIV II Panel 9-82, Analog Trip Units Connection Diagrams SH-1, Rev. 14
2-730E928-2, Elementary Diagram HPCI System, Rev. 25
2-730E928-3, Elementary Diagram HPCI System, Rev. 22
2-730E928-4, Elementary Diagram HPCI System, Rev. 10
2-730E928-5, Elementary Diagram HPCI System, Rev. 27

Other Documents
BFN-VTD-A348-0200, AMERACE Agastat Product Specifications for Model ETR Series Time Delay Relay Class 1E, Rev. 2
CDQ000999201200068, Seismic Evaluation of Mounting Details for Components Installed in Panels 925-0048A, B, C and D, Rev. 1
CR 1263268
DCN 70563
Equipment Apparent Cause Evaluation Report for CR 1263268, Rev. 1

Section 4OA5
Drawings
8706-1, Assembly, Lift Yoke, HI-TRAC VW, Ancillary #702, Rev. 2
8706-2, Assembly, Lift Yoke, HI-TRAC VW, Ancillary #702, Rev. 2

Procedures
MSI-0-000-LFT001, Lifting Instructions for the Control of Heavy Loads, Rev. 72
MSI-0-079-DCS035, Dry Cask Campaign Guidelines, Rev. 19
MSI-0-079-DCS036(FW), ISFSI Abnormal Conditions Procedure for HI-STORM (FW) System, Rev. 0
MSI-0-079-DCS200.2, MPC-68 Loading and Transport Operations, Rev. 30
MSI-0-079-DCS400.1FW, ISFSI Abnormal Conditions Procedure Placing the MPC-89 in a Safe Condition, Rev. 1

<u>Other Documents</u>

CDQ007920050039, Structural Analysis of HI-TRAC VW Lift Yoke, Rev. 1
Certificate of Compliance for Spent Fuel Storage Casks, Amendment 5
Holtec Report No. HI-2114830, Final Safety Analysis Report on the Hi-Storm FW MPC Storage System, Rev. 2.1a
DCN 70980, Rev. A