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

February 1, 2016

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000259/2015007, 05000260/2015007, AND 05000296/2015007

Dear Mr. Shea:

On December 18, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3 and discussed the results of this inspection with Mr. S. Bono and other members of your staff. Additional inspection results were discussed with Mr. L. Hughes and other members of the licensee's staff on January 21, 2016. Inspectors documented the results of this inspection in the enclosed inspection report.

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

NRC inspectors documented two findings of very low safety significance (Green) in this report. These findings involved violations of NRC requirements. The NRC is treating these violations as a non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest 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 the Browns Ferry Nuclear Plant.

If you disagree with the cross-cutting aspect assigned 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 Regional Administrator, Region II, and the NRC resident inspector at the Browns Ferry Nuclear Plant.

In accordance with Title 10 of the Code of Federal Regulations 2.390, “Public Inspections, Exemptions, Requests for Withholding,” 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 NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000259/2015007, 05000260/2015007 and 05000296/2015007
w/ Attachment: Supplementary Information

cc: Distribution via Listserv

In accordance with Title 10 of the Code of Federal Regulations 2.390, "Public Inspections, Exemptions, Requests for Withholding," 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 NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000259/2015007, 05000260/2015007 and 05000296/2015007
w/ Attachment: Supplementary Information

cc: Distribution via Listserv

☒ PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE ☒ NON-SENSITIVE
ADAMS:☒ Yes ACCESSION NUMBER:______________ ☒ SUNSI REVIEW COMPLETE ☒ FORM 665 ATTACHED

OFFICE	RII:DRS	RII:DRP	RII:DRS	RII:DCI	RII:DRS	RII:DRS	RII:DRP
SIGNATURE	**GKO**	**WXD1**	**RNP1**	**DXT2**	**MXY4**	**JXC22**	**AJB3**
NAME	G. Ottenberg	W. Deschaine	R. Patterson	D. Terry-Ward	M. Yeminy	J. Chiloyan	A. Blamey
DATE	01/22/2016	01/ /2016	01/27/2016	01/25/2016	01/23/2016	01/22/2016	01/29/2016
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
OFFICE	RII:DRS						
SIGNATURE	**JHB1**						
NAME	J. Bartley						
DATE	2/1/2016						
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: S:\DRS\ENG BRANCH 1\BRANCH INSPECTION FILES\2014-2015-2016 CYCLE INSPECTION FOLDER FOR ALL SITES\BROWNS FERRY\2015 CDBI\BFN CDBI REPORT 2015-007 FINAL.DOCX

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report Nos.:	05000259/2015007, 05000260/2015007 and 05000296/2015007
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Corner of Shaw and Nuclear Plant Road Athens, AL 35611
Dates:	November 16, 2015 – December 18, 2015
Inspectors:	W. Deschaine, Resident Inspector (Lead) G. Ottenberg, Senior Reactor Inspector D. Terry-Ward, Construction Inspector R. Patterson, Reactor Inspector J. Chiloyan (Electrical) M. Yeminy (Mechanical)
Approved by:	Jonathan Bartley, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY

IR 05000259/2015007, 05000260/2015007, 05000296/2015007; 11/16/2015 – 12/18/2015; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Component Design Bases Inspection.

This inspection was conducted by a team of three Nuclear Regulatory Commission (NRC) inspectors from Region II, one resident inspector, and two NRC contract personnel. Two Green non-cited violations (NCVs) were identified. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using the NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5, dated February 2014.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. A NRC-identified non-cited violation (NCV) of Technical Specifications (TS) 5.4.1 was identified for the failure to develop a preventive maintenance (PM) schedule that specified inspection of the Emergency Diesel Generators (EDG) neutral grounding resistor as recommended by Regulatory Guide (RG) 1.33, 9.b. Specifically, procedures failed to provide proper guidance to maintain the grounding resistor in accordance with design basis as described in the UFSAR and electrical calculations. Upon identification of the issue, the licensee performed a visual inspection of the resistor and determined that it was functional based on no signs of physical degradation or damage. The licensee entered this issue into the corrective action program (CAP) as CR1114779 to evaluate and implement appropriate corrective actions.

This performance deficiency was more than minor because if left uncorrected it could result in a more significant safety concern. Specifically, lack of inspections of the secondary grounding resistor could allow for an undetected condition which would cause transient voltages capable of damaging safety related equipment. The finding was screened for significance using the Mitigating Systems cornerstone column of IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," dated June 19, 2012, and was determined to be of very low safety significance (Green) using IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, because the finding affected the design or qualification of a Mitigating SSC, and the SSC maintained its operability as documented in CR 1114779. No cross-cutting was assigned because it is not indicative of current licensee performance. (Section 1R21.2)

Cornerstone: Barrier Integrity

Green. A NRC identified NCV of 10 Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion XI, "Test Control," was identified for the failure to specify adequate test instrumentation for performing MSIV leak rate testing. Specifically, the licensee test procedure allowed the use of high range test instruments to measure low leakage rates while performing the combined leak rate testing on the Unit 1 B Main Steam Line. This resulted in instrument uncertainties large enough to impact the validity of the test results.

The licensee immediately entered this issue into their corrective action program as CR 1117381. The licensee performed an evaluation and determined that the latest test results provided reasonable assurance of operability.

This performance deficiency was more than minor because if left uncorrected had the potential to lead to a more significant safety concern by masking the failure to meet test acceptance criteria. The finding was screened for significance using the Barrier Integrity cornerstone column of IMC 0609, Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” dated 7/1/2012, and IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated 7/1/2012, and was determined to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment. This finding was assigned a cross-cutting aspect in the area of Problem Identification and Resolution because the licensee did not initiate a corrective action to identify the cause of the negative leak rate results obtained during the recent performance of the test procedure (P.1). (Section 1R21.2)

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program (CAP). That violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21M)

.1 Inspection Sample Selection Process

The team selected risk-significant components and related operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1E-6. The sample included 12 components, two of which were associated with containment large early release frequency (LERF), and two operating experience (OE) items.

The team performed a margin assessment and a detailed review of the selected risk-significant components and associated operator actions to verify that the design bases had been correctly implemented and maintained. Where possible, this margin was determined by the review of the design basis and Updated Final Safety Analysis Report (UFSAR) response times associated with operator actions. This margin assessment also considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Manual Chapter 0326 conditions, NRC resident inspector input regarding problem equipment, system health reports, industry OE, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, OE, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Component Reviews

a. Inspection Scope

Components

- Unit 2 Main Bank Battery (B)
- 480 Volt (V) Alternating Current (AC) Reactor Motor Operated Valve (RMOV) Board 1B
- 250 V RMOV Board 1A
- Units 1 and 2 Emergency Diesel Generator (EDG) B
- Units 1 and 2 4160V Shutdown Board A
- C1 Residual Heat Removal Service Water (RHRSW) Pump
- Unit 1 High Pressure Coolant Injection (HPCI) pump
- Unit 2 HPCI System Inboard Isolation Motor-Operated Valve (MOV)
- Unit 1 Residual Heat Removal (RHR) Loop II Pump Test Return MOV
- Emergency Equipment Cooling Water (EECW) System North and South Header Check Valves

- Unit 1 & 2 EDG Ventilation System
- Unit 1 C RHR Heat Exchanger

<u>Components with LERF Implications</u>

- Unit 1 Main Steam Isolation Valves (MSIV) (15, 27, 38, 52)
- Unit 1 Feedwater Check Valves (3-558, 572, 554, 568)

For the 14 components listed above, the team reviewed the plant technical specifications (TS), UFSAR, design bases documents (DBDs), and drawings to establish an overall understanding of the design bases of the components. Applicable design calculations and procedures were reviewed to verify that the design and licensing bases had been appropriately translated into these documents. Test procedures and recent test results were reviewed against DBDs to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents, and that individual tests and analyses served to validate component operation under accident conditions. Maintenance procedures were reviewed to ensure components were appropriately included in the licensee's preventive maintenance program. System modifications, vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action program documents were reviewed (as applicable) in order to verify that the performance capability of the component was not negatively impacted, and that potential degradation was monitored or prevented. Maintenance Rule information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current Maintenance Rule status. Component walkdowns and interviews were conducted to verify that the installed configurations would support their design and licensing bases functions under accident conditions and had been maintained to be consistent with design assumptions.

Additionally, the team performed the following component-specific reviews:

- The team observed a simulator scenario involving the alignment of one train of the RHR system into the suppression pool cooling mode of operation following a loss of coolant accident to verify the design conditions assumed for the RHR loop II pump test return MOV were in alignment with the conditions imposed by the operating procedure.
- The team reviewed the most recent MOV diagnostic testing results for the RHR loop II pump test return MOV and the HPCI inboard isolation MOV to verify current MOV parameters are bounded by their requirements and capability assumptions.
- The team observed EECW system and check valve testing on 12/1/15, to verify appropriate system configurations were used to confirm the EECW north and south header check valves were meeting American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code testing requirements.
- The team verified the licensee was testing the EECW check valves in accordance with their ASME OM Code of record requirements for the check valve condition monitoring program.
- The team observed a performance test of the 1C RHR heat exchanger.
- The team reviewed photographs of macro fouling (clam shells) found in March 2015 inside the 2C heat exchanger.

- The team reviewed the validity of an uncontrolled MathCad analysis of heat exchanger test results which was used to determine Past Operability of the 2C RHR heat exchanger, including a review of a missing MathCad file that had to be resurrected during the inspection.
- The team reviewed the adequacy of a third party (Zachry) analysis of the as tested fouling factors associated with the 2A and 2C RHR heat exchangers and assessed the magnitude of the error inherent in incorrect input of the number of plugged tubes.
- The team reviewed the validity of methodology, assumptions and values used in 10 CFR 50, Appendix J testing.
- The team reviewed the data used to ascertain that the Feedwater containment isolation valves have passed their inservice testing acceptance criteria.
- The team performed independent calculations of available fault current contributions from the emergency diesel generator and from the offsite sources for postulated phase and ground faults and compared them with the relay settings calculations in Electrical Transient Analysis Program (ETAP) to verify the appropriateness of the applied overcurrent relay settings.
- The team reviewed Shutdown Board A loss of voltage and bus overcurrent relay settings to ensure adequate coordination was maintained between the bus overcurrent and bus under voltage relay settings to ensure the overcurrent relays function as designed during postulated electrical bus faults.
- The team reviewed the degraded voltage relay settings to verify whether they bounded the TS requirements.
- Protective relay setpoint calculations and setpoint calibration test results were reviewed to assess the adequacy of protection during testing and emergency operations.
- The permissive and interlocks associated with the EDG B output breaker were reviewed to determine whether the breaker opening and closing control circuits were consistent with design basis documents.
- The team selectively reviewed board and breaker ratings, load flow calculations, degraded voltage calculations, and protective device settings, to confirm that the MOV board would be capable of supplying the necessary loads for mitigating design bases events and for achieving safe shutdown in accordance with the design bases.
- The team selectively reviewed electrical one-line diagrams, loading calculations, voltage drop calculations, short-circuit calculations, and associated electrical protection to determine the capability of the 250 V direct current (DC) RMOV Board 1A to serve the required power to 250 VDC downstream loads in accordance with the design and licensing basis as well as for station blackout events.
- Battery sizing, loading, and voltage calculations were reviewed, as were maintenance and operational procedures, in order to verify that design bases and design assumptions have been appropriately translated into design calculations and procedures.
- Battery room temperature and ventilation for hydrogen gas during charging evaluations were reviewed to verify that the equipment qualification was suitable for the environment expected under all conditions.

b. Findings

.1 Failure to Specify Adequate Instrument Ranges for MSIV Leakage Testing

Introduction: A Green, NRC-identified non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” was identified for the failure to specify adequate test instrumentation for performing MSIV leak rate testing. Specifically, the licensee test procedure allowed the use of high range test instruments to measure low leakage rates while performing the combined leak rate testing on the Unit 1 B Main Steam Line. This resulted in instrument uncertainties large enough to impact the validity of the test results.

Description: On October 25, 2014, the licensee performed MSIV combined leak rate testing in accordance with procedure 1-SR-3.6.1.3.10(B), “Primary Containment Local Leak Rate Test Main Steam Line B: Penetration X-7B,” revision 4, to satisfy the requirements in the sites Containment Leak Rate Program. 10 CFR 50, Appendix B, Criterion XI required that the licensee establish a test program to ensure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was identified and performed in accordance with written test procedures and that test procedures include provisions for assuring that adequate test instrumentation was available and used.

The primary function of the MSIVs is to prevent damage to the fuel barrier and limit release of radioactive materials by closing the primary containment barrier. This local leak rate test (LLRT) was performed to quantify the leakage through Main Steam Line “B” primary containment isolation valves 1-FCV-1-26 and 1-FCV-1-27 and demonstrate the following:

1. Leakage through 1-FCV-1-26 and 1-FCV-1-27 was within the limit specified by Technical Specification Surveillance Requirement 3.6.1.3.10.

2. The combined main steam line path leak rate was within the limits specified by Technical Specification SR 3.6.1.3.10.

Leak rates obtained by the performance of this procedure were tabulated to demonstrate whether or not the combined main steam line path leak rate was within the limits specified in the technical specification when primary containment is being established or maintained. The team identified that the “B” Main Steam Isolation Valve testing indicated that the inboard valve had a negative leak rate which wasn’t possible. For the test results the licensee evaluated the negative leak rates as being zero leakage. The licensee did not initiate a corrective action to evaluate the cause of the negative leakage rate test results. During the team’s review of the tabulated test results, the team questioned whether or not the licensee considered instrument uncertainties when evaluating test results. The team identified that, while performing the combined leakage test on the B Main Steam Line, a 0-800 standard cubic feet per hour (scfh) gauge was used to measure a leak rate of 25 scfh and the test evaluator did not account for the +/- 2% (16scfh) error associated with the test instrument when evaluating the test results. Using larger range instruments yield higher uncertainties and present testing results that can potentially mask the failure to meet the test acceptance criteria. The team concluded that the licensee failed to establish adequate measures to ensure that testing equipment used to perform combined leak rate test was appropriate for the application and therefore, had adversely affected the ability to accurately characterize the as-found condition. The licensee immediately entered this issue into their corrective action program as CR 1117381. The licensee performed an evaluation and determined that the latest test results provided reasonable assurance of operability.

Analysis: The licensee's failure to specify adequate instrument test ranges was a performance deficiency. Specifically, while performing the combined leakage test on the B Main Steam Line a 0-800 scfh gauge was used to measure leak rate of 25 scfh. The significant uncertainty of the gauge could mask the failure to meet the test acceptance criteria. This performance deficiency was more than minor because if left uncorrected had the potential to lead to a more significant safety concern by masking the failure to meet test acceptance criteria. The finding was screened for significance using the Barrier Integrity cornerstone column of IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," dated 7/1/2012, and IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated 7/1/2012, and was determined to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment. This finding was assigned a cross-cutting aspect in the area of Problem Identification and Resolution because the licensee did not initiate a correct action to identify the cause of the negative leak rate results obtained during the recent performance of the test procedure (P.1).

Enforcement: 10 CFR 50, Appendix B, Criterion XI, "Test Control" requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures and that test procedures include provisions for assuring that adequate test instrumentation is available and used. Contrary to the above, on July 16, 2014, the licensee did not establish written procedures that included provisions to assure that adequate test instrumentation was used. Specifically, MSIV LLRT test procedures did not specify the appropriate range of test instrumentation to ensure instrument uncertainty did not mask the failure to meet test acceptance criteria. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensees CAP as CR 1117381. (NCV 05000259, 260, 296/2015007: Failure to Specify Adequate Instrument Ranges for MSIV Leakage Testing)

.2 Failure to develop a PM schedule that specified inspection of the EDG neutral grounding resistor

Introduction: A Green NRC-identified non-cited violation (NCV) of TS 5.4.1 was identified for the licensee's failure to develop a PM schedule that specified inspection of the Emergency Diesel Generators (EDG) neutral grounding resistor as recommended by RG 1.33, 9.b. Specifically, procedures failed to provide proper guidance to maintain the grounding resistor in accordance with design basis as described in the UFSAR and electrical calculations.

Description: As described in the BFN UFSAR Section 8.5.3.2, each diesel generator is wye-connected with its neutral grounded through a distribution transformer and secondary resistor. This method of grounding is to provide a high-resistance grounding system to limit ground fault current contribution from the EDGs for postulated ground faults on the 4160V safe shutdown boards and associated feeders. The ground fault current is limited to a few amperes to minimize equipment damage due to transient overvoltages present in an ungrounded system, and to allow uninterrupted operation of EDG and connected loads under the presence of a single fault to ground. In the event the secondary resistor was degraded to the point that it would look like an open circuit the EDG would become an ungrounded system, and it may not reveal its condition during routine surveillance testing. This would allow transient voltages to exist on the system that would be capable of damaging safety related equipment. A failure of this resistor could go undetected and result in damage to safety related equipment.

Technical Specifications 5.4.1, Procedures, requires the licensee to establish implement and maintain the applicable procedures recommended in Regulatory Guide (RG) 1.33. Regulatory Guide 1.33, Section 9.b states that preventative maintenance schedules be developed to specify inspections of equipment. The team identified that the licensee had not implemented preventative maintenance (PM) schedules to inspect the secondary resistor for degradation that could impact its design function. Upon the team's identification of the issue, the licensee performed a visual inspection of the resistor and determined that it was functional based on no signs of physical degradation or damage. The licensee entered this issue into the corrective action program (CAP) as CR1114779 to evaluate and implement appropriate corrective actions.

<u>Analysis</u>: The licensee's failure to develop a PM schedule that specified inspection of the EDG neutral grounding resistor as recommended by RG 1.33, 9.b was a performance deficiency. This performance deficiency was more than minor because if left uncorrected it could result in a more significant safety concern. Specifically, lack of inspections of the secondary grounding resistor could allow for an undetected condition which would cause transient voltages capable of damaging safety related equipment. The finding was screened for significance using the Mitigating Systems cornerstone column of IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," dated June 19, 2012, and was determined to be of very low safety significance (Green) using IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, because the finding affected the design or qualification of a Mitigating SSC, and the SSC maintained its operability as documented in CR 1114779. No cross-cutting was assigned because it is not indicative of current licensee performance.

<u>Enforcement</u>: Technical Specification 5.4.1 required that, written procedures shall be established, implemented, and maintained covering applicable procedures recommended in Regulatory Guide 1.33, revision 2, appendix A, February 1978. Regulatory Guide 1.33, appendix A, Section 9.b states, in part, that preventive maintenance schedules should be developed to specify inspections of equipment. Contrary to the above, since 1995 the licensee failed develop a PM schedule that specified inspection of the EDG neutral grounding resistor to assure that the EDG grounding resistor would be able to maintain its safety function and ensure the operability of safety-related EDGs during a design basis event. Upon the team's identification of the issue, the licensee performed a visual inspection of the grounding resistor to verify its current functionality and entered the issue in their CAP to evaluate and implement appropriate monitoring measures. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensees CAP as CR 1114779. (NCV 05000259, 260, 296/2015007-02, Failure to develop a PM schedule that specified inspection of the EDG neutral grounding resistor)

.3 <u>(Opened) Procurement of Electrical Equipment for Ungrounded Electrical Systems</u>

Introduction: The team identified an unresolved item (URI) related to the licensee's procurement of electrical equipment for ungrounded electrical systems.

Description: The 480 VAC system for each unit consists of 480-V Load Center Unit Substations with each substation consisting of 4160-480-V transformers, primary terminal box, and close-coupled or bus duct connected 480-V, metal-enclosed switchgear. The 480-VAC distribution system is three-phase ungrounded. Each substation bus is normally fed from its own transformer, with an alternate source consisting either of an adjacent 480-VAC bus section or of another transformer serving as standby. Ventilated dry-type transformers are three-phase, delta-delta configuration so that the 480 VAC system is ungrounded.

Ungrounded systems are susceptible to overvoltage conditions resulting from a single line to ground fault. A line to ground fault will result in a sustained higher voltage to ground on the ungrounded phases. Industry standard IEEE 242 (Buff Book) "Protection and Coordination of Industrial and Commercial Power Systems", section 8.2.5 "Ungrounded Systems" stated "if this ground fault is intermittent or allowed to continue, the system could be subjected to possible severe overvoltages to ground, which can be high"…(cause line to ground voltages several times normal voltage on all three phases). Because of the potential for overvoltage conditions, specifications, purchase orders or procurement documents for equipment such as motors, cables, and switchgear should identify that the equipment is intended for use on an ungrounded system.

The team requested the original specifications for the installed BFN safety-related motors BFN-2-MTR-068-0003; 2-FCV-68-3 (Recirc Pump 2A Disch VLV) fed from the 480V Reactor MOV Board 2E and, BFN-3-MTR-073-0002; 3-FCV-73-2 (HPCI Steam Line INBD Isolation VLV) fed from the 480V Reactor MOV Board 3A to determine if the intended service condition as a 480 VAC ungrounded system was appropriately identified. The team reviewed Procurement Engineering Group packages CRP205J - PO 733602 and CFK570P – PO 836093 for the safety-related motors and determined that the ungrounded system requirement was not identified.

Equipment intended for service on ungrounded systems is designed to withstand the sustained higher line to ground voltages than can occur on grounded systems. These insulation systems are not typically provided unless the purchaser specifies an ungrounded system. Industry standard "NEMA MG 1 "Motors and Generators", section 14.31 "Machines Operating On An Ungrounded System" stated:

> "Alternating-current machines are intended for continuous operation with the neutral at or near ground potential. Operation on ungrounded systems with one line at ground potential should be done only for infrequent periods of short duration, for example as required for normal fault clearance. If it is intended to operate the machine continuously or for prolonged periods in such conditions, a special machine with a level of insulation suitable for such operation is required. The motor manufacturer should be consulted before selecting a motor for such an application."

The NRC will review the licensee's responses to follow-up questions asked during a conference call with the licensee on January 21, 2016. Based on this future review, the NRC will make a determination if the licensee properly procured electrical components for ungrounded systems. More information is needed to determine if more than a minor performance deficiency or violation exists associated with this issue, thus a URI is being opened. (URI 05000259, 260, 296/2015007-03, Procurement of Electrical Equipment for Ungrounded Electrical Systems).

.3 Operating Experience

a. Inspection Scope

The team reviewed two operating experience issues for applicability at the Browns Ferry Nuclear Power Station. The team performed an independent review of these issues and, where applicable, assessed the licensee's evaluation and dispositioning of each item. The issues that received a detailed review by the team included:

- NRC Information Notice (IN) 2008-20, Failures of Motor Operated Valve Actuator Motors with Magnesium Alloy Rotors
- NRC IN 90-43, Mechanical Interference with Thermal Trip Function in GE Molded-Case Circuit Breakers

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On December 18, 2015, the team presented the inspection results to Mr. S. Bono and other members of the licensee's staff. Additional inspection results were discussed with Mr. L. Hughes and other members of the licensee's staff on January 21, 2016. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

4OA7 Licensee-identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

Technical Specification 5.4.1, required, in part, that "written procedures shall be established, implemented, and maintained covering the following activities: a. the applicable procedures recommended in Regulatory Guide (RG) 1.33, Rev 2, Appendix A." Procedures recommended in Appendix A to RG 1.33, included procedures for performing maintenance, and specifically, "preventive maintenance schedules should be developed to specify…inspections of equipment." 0-TI-522, "Program for Implementing NRC Generic Letter 89-13" required in part, in section 7.2, that "BFN will maintain an inspection and cleaning program in accordance with the BFN PM Program to verify the heat transfer capability of the safety related Heat Exchangers cooled by EECW an RHRSW," and "the PMs provide for reassessing this inspection frequency based on the results of inspections, not to exceed 5 years." Contrary to this requirement, since January 22, 2013, the licensee did not implement their PM schedule for inspections of the 3C RHR HX appropriately, because they allowed the PM to extend beyond the maximum of 5 years. Consequently, when the heat exchanger was opened, it failed the acceptance criterion of no more than 77 tubes plugged. This finding was not greater than very low safety significance (Green) because it was a deficiency affecting the design of a Mitigating SSC, and the SSC maintained its operability or functionality (as demonstrated by past operability evaluation for PERs 750848 and 750858). The licensee entered this issue into their CAP as CR 674040.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

<u>Licensee personnel:</u>
S. Bono, Site Vice President
L. Hughes, General Plant Manager
P. Summers, Director of Safety and Licensing
J. Paul, Nuclear Site Licensing Manager
M. McAndrew, Manager of Operations
D. Campbell, Superintendent of Operations
M. Kirschenheiter, Assistant Director for Site Engineering
K. Groom, Design Engineering Manager
R. Jarrett, Corp Design
P. Wilson, Corp Licensing
T. Scott, QA Manager
W. Anderson, Plant Support
P. Derriso, Engineering
M. Acker, Licensing Engineer
T. Cole, Radiation Protection
J. Smith, System Engineer
P. Campbell, System Engineer
K. Skinner, System Engineer
L. Holland, System Engineer
D. Ford, System Engineer
D. Jackson, RHRSW/EECW System Engineer
J. Lacasse, RHR System Engineer
C. McDonald, Site MOV Program Engineer
E. Ridgell, IST Program Engineer

<u>NRC personnel</u>
M. Farnan, Mechanical Engineer, Office of Nuclear Reactor Regulation (NRR)

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

<u>Opened and Closed</u>

05000259,260,296/2015007-01:	NCV	Failure to Specify Adequate Instrument Ranges for MSIV Leakage Testing [Section 1R21.2]
05000259,260,296/2015007-02:	NCV	Failure to develop a PM schedule that specified inspection of the EDG neutral grounding resistor [Section 1R21.2]

<u>Opened</u>

05000259,260,296/2015007-03:	URI	Procurement of Electrical Equipment for Ungrounded Electrical Systems [Section 1R21.2]

LIST OF DOCUMENTS REVIEWED

Procedures

NPG-SPP-01.2, Administration of Site Technical Procedures, Revision 12
OPL171.026, Reactor Feedwater System, Revision 15
NEDP-27, Past Operability Evaluations, Revision 3
NPG-SPP-22.302, Corrective Action Program Screening, Revision 6
NPDP-8, Operability Determination Process and Limiting Conditions for Operation Tracking, Revision 18
Primary Containment Local Leak Rate Test Reactor Feedwater Line B: Penetration X-9B
1-SI-4.7.A.2.g-3/3a, Primary Containment Local Leak Rate Test Reactor Feedwater Line A: Penetration X-9A, Revision 7
0-TI-426, Diesel Generator Exhaust Fan Flow Verification, Revision 11
0-SR-3.8.1.1(B) Diesel Generator Bi Monthly Operability Test, Revision 59
0-TI-322, RHR Heat Exchanger Performance Testing, Revision0
0-TI-63, RHRSW Flow Blockage Monitoring, Revision 27
NGP-SPP-09.14, Generic Letter (GL) 89-13 Implementation, Revision 3
N0301, Zachry Engineering Calculations, Revision 2
0-TI-522, Program for Implementing NRC Generic Letter 89-13, Revision 5
MMTP-102, Erection of Scaffolds/Temporary Work Platforms and Ladders, Revision 11
0-TI-360, Containment Leak rate Programs, Revision 40
0-TI-577(TEST), In-service testing of ASME and Augmented Pressure Relief Devices, Revision 7
0-TI-579, RHRSW System Pump Baseline Data Evaluation, Revision 5
1-SR-3.3.1.1.14(5II), MSIV RPS Trip Logic System Functional Test (Channel B1/B2), Revision 2
PMTI-70491-004, Installation of Interposing Lockout Relay and Indicating Light in Dedicated Fused Circuit, Revision0
MMTP-104, Guidelines and Methodology for Assembling and tensioning Threaded Connections, Revision 0007
NEDP-2, Design Calculation Process Control, Revision 0017
NEDP-8.2, Technical Evaluation for Procurement of Safety Related and Quality Related Materials, Items, and Services, Revision 0001
NEDP-8.4, Equivalency Evaluation for Procurement and Use of Replacement Materials and Items, Revision 0001
NPG-SPP-22.300, Corrective Action Program, Revision 003
TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan (NQAP) (Quality Assurance Program Description), Revision 0028
1-SR-3.8.4.3(MB-1), Surveillance Procedure, Main Bank 1 Battery Service Test, Revision 0007
1-SR-3.8.4.1(1), Weekly Check for 250V Main Bank Number 1 Battery, Revision 0016
1-SR-3.8.6.2(1), Quarterly Check of 250 Volt Main Bank Number 1 Battery, Revision 0015

Completed Procedures

0-TI-545, EECW System Individual Load Flow Measurements and Adjustments, Revision 1, dated 3/1/08
1-SI-4.7.A.2.G-3/74B, Primary Containment Local Leak Rate Test RHR Suppression Pool Spray: Penetration X-211A, Revision 2, dated 10/31/08
1-SI-4.7.A.2.G-3/74E, Primary Containment Local Leak Rate Test RHR Suppression Pool Spray: Penetration X-211B, Revision 2, dated 11/3/08
2-SR-3.5.1.1, Maintenance of Filled HPCI Discharge Piping, dated 9/15/2011
2-SR-3.5.1.1, Maintenance of Filled HPCI Discharge Piping, dated 10/18/2011
2-SR-3.5.1.1, Maintenance of Filled HPCI Discharge Piping, dated 11/16/2011

2-SR-3.5.1.1, Maintenance of Filled HPCI Discharge Piping, dated 12/20/2011
1-SR-3.5.1.7, HPCI Comprehensive Pump Test, dated 1/25/2015
1-SI-3.2.12, Outboard MSIV Fail Safe Test, dated 10/8/2015
1-SI-5.5.6, MSIV Stroke Time Test, dated 10/4/2015
1-SR-3.3.1.1.14 (5I), MSIV RPS Trip Logic Test, dated 2/28/2015
0-TI-383, Evaluation of Test Results for the ASME OM Code IST Test Results, dated 8/31/2015
1-SR-3.6.1.3.10(D-OUTBD), LLRT Main Steam Line D Outboard Penetration, dated 10/3/2014
1-SR-3.6.1.3.10(D-OUTBD), LLRT Main Steam Line D Outboard Penetration, dated 10/6/2014
1-SR-3.6.1.3.10(D-OUTBD), LLRT Main Steam Line D Outboard Penetration, dated 10/23/2014
1-SR-3.6.1.3.10(D), COMBINED LLRT Main Steam Line D Outboard Penetration, dated 10/6/2014
1-SR-3.6.1.3.10(C-OUTBD), LLRT Main Steam Line C Outboard Penetration, dated 10/3/2014
1-SR-3.6.1.3.10(C), LLRT Main Steam Line C Outboard Penetration, 10/6/2014
1-SR-3.6.1.3.10(B-OUTBD), LLRT Main Steam Line B Outboard Penetration, dated 10/3/2014
1-SR-3.6.1.3.10(B-OUTBD), LLRT Main Steam Line B Outboard Penetration, dated 10/4/2014
1-SR-3.6.1.3.10(B-OUTBD), LLRT Main Steam Line B Outboard Penetration, dated 10/5/2014
1-SR-3.6.1.3.10(B-OUTBD), LLRT Main Steam Line B Outboard Penetration, dated 10/23/2014
1-SR-3.6.1.3.10(B), COMBINED LLRT Main Steam Line B Outboard Penetration, dated 10/6/2014
1-SR-3.6.1.3.10(A-OUTBD), LLRT Main Steam Line A Outboard Penetration, dated 10/3/2014
1-SR-3.6.1.3.10(A-OUTBD), LLRT Main Steam Line A Outboard Penetration, dated 10/5/2014
1-SR-3.6.1.3.10(A-OUTBD), LLRT Main Steam Line A Outboard Penetration, dated 10/23/2014
1-SR-3.6.1.3.10(A), LLRT Main Steam Line A Outboard Penetration, dated 10/6/2014
1-SR-3.6.1.3.10(OPT-A), LLRT Main Steam Line A Outboard Penetration, dated 10/28/2014
0-SI-4.5.C.1(C1-COMP), RHRSW Pump C1 IST Comprehensive Pump Test, dated 4/2/2014
0-SI-4.5.C.1(C1), RHRSW Pump C1 IST Group A Pump Test, dated 4/10/2014
1/2-SI-4.9.A.1.d(B), Diesel Generator B 2 Year Inspection, dated 10/9/13
0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, dated 11/22/15
0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, dated 10/25//15
0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, dated 9/27/15

Modifications
DCN 71313, Replacement of All RHRSW Pumps, Revision A
PMTI-66314, MSIV Poppet Modification Project

Drawings
0-A-12335-M-1-E, Cast Steel Globe Valve with Limitorque SMB-2 Actuator, Revision 3
1-45E779-23, Wiring Diagram 480V Shutdown Aux Power Schematic Diagram, Revision 7
1-47A370-74-73, Mechanical Limit Switch Development and Valve Thrust Requirements, Revision 4
1-47B370-2, Mechanical Motor operated Valves Testing Requirements, Revision 7
1-47BD452-B4, 1-MVOP-074-0073 Operator Data Sheet, Revision 2
1-47E408-1, Mechanical Valve Actuator (EMO) Modifications, Revision 3
1-47E408-2, Mechanical Motor operated Valves Internal Dimensions for PPM, Revision 3
1-47E811-1, Flow Diagram Residual Heat Removal System, Revision 45
2-47E408-1, Mechanical Valve Actuator (EMO) Modifications, Revision 17
2-47E408-2, Mechanical Motor operated Valves Testing Requirements, Revision 17
2-47E812-1, Flow Diagram High Pressure Coolant Injection System, Revision 71
PC-139988, 14" L-900 W.E.O.S Press. Seal Gate Valve with SMB-4T Limit. Unit & Lantern Gland, Revision 1
1-47E811-1, Flow Diagram Residual Heat Removal System, Revision 45

1-47E858-1, RHRSW Flow Diagram, Revision 72
0-37W205-5, Mechanical Pumping Station & Water Treatment Drawing, Revision 8
1-47E812-1, HPIC System Flow Diagram, Revision 43
1-730E915, MSIV Logic Diagram Sheet 9, Revision 20
1-730E915, MSIV Logic Diagram Sheet 10, Revision 21
1-730E915RE, MSIV Logic Diagram Sheet 12, Revision 5
1-47E801-1, Mechanical Flow Diagram Main Steam, Revision 24
0-45E765-9, 4160V Shutdown Auxiliary Power Schematic Diagram, Revision 1
0-45E767-3, Wiring Diagram Diesel Generators Schematic Diagram, Revision 20
0-45E767-3, Wiring Diagram Diesel Generators A-D Schematic Diagram, Revision 15
0-731E718-1, Elementary Diagram Diesel Generator Prot Relaying & Metering, Revision 14
0-731E761-4, Elementary Diagram Emergency Equipment, Revision11
0-731E747-1, One Line Diagram Diesel Generator Panel, Revision 6
0-45E724-2, Wiring Diagram 4160V Shutdown BD B Single Line, Revision 42
0-15E500-1, Key Diagram of Standby Auxiliary Power System, Revision 43
0-15E500-1, Key Diagram of Standby Auxiliary Power System, Revision 043
0-15E500-2, Key Diagram of Normal Auxiliary Power System, Revision 027
0-15E500-3, Key Diagram of Normal & Standby Auxiliary Power System, Revision 068
0-45N230 Electrical Equipment Battery & DC EQPT Rooms Plans, Sections & Details, Revision 005
0-45N233 Electrical Equipment Battery & DC EQPT Room Details & Sections, Revision 003
0-45N382, Electrical Equipment 250V DC Reactor MOV BD 1A, 2A, 3A Outline & General Arrangement, Revision 006
0-47E605-1, Mechanical Layout of Control Boards, Revision 033
0-45N700-1, Switchboards Battery Board 1, Front Elevation, Revision 004
0-47E200-5, Equipment Plans – El 593.0 & 586.0, Revision 020
0-45E701-1 Wiring Diagram, Battery BD 1, Panels 1-7, Single Line, Revision 069
0-45E702-1 Wiring Diagram, Battery Board 2, Panel 1-7, Single Line, Revision 064
0-45E710-2 Wiring Diagram, Inst & Controls, DC & AC Power System Key Diagram, Revision 018
1-45E751-3 Wiring Diagram, 480V Reactor MOV BD 1B, Single Line, Revision 037
1-45E751-4 Wiring Diagram, 480V Reactor MOV BD 1B, Single Line, Revision 045
1-45E712-1, Wiring Diagram 250V Reactor MOV D 1A, Single Line, Revision 033
1-45E712-2, Wiring Diagram 250V Reactor MOV D 1B, Single Line, Revision 041
1-45E714-7, Wiring Diagram 250V DC Reactor MOV BDS Schematic Diagram, Revision 005
1-45N806-1, Conduit & Grounding, Floor EL 593.0 Plans Revision 004
2-45N806-2, Conduit & Grounding, Floor EL 593.0 Plans, Revision 002
1-M-5886-2, Miscellaneous Steel Battery Rack Layout, Revision 000
1-M-5886-2, Miscellaneous Steel Battery Rack Layout, Revision 001
1-45N1771-8, Wiring Diagrams, 250V DC Reactor MOV BD 1A, Connection Diagram, R000
1-45N1711-6, Wiring Diagrams, 250V D-C Reactor MOV BD 1A, Connection Diagram SH-6, R000

Calculations
351401-BFN, Thrust Capacity Calculation Weak Link, CQV002B, CQF814B, CQF818Y, CQV001B, CQF828V, WL-082 Thrust Capacity Calculation, Revision 0
EDQ0057920034, 4.16KV and 480V Busload, Voltage Drop and Short Circuit Calculation, Revision 98
EDQ024820020042, 250V DC Unit Batt Load Study, VD, SC, and Batt Capacity for LOCA/LOOP, Station Blackout and Appendix R Analysis for Unit / Shutdown Board Battery, Revision 69

MDQ0031920537, HVAC Adequacy Analysis- Control Bay, Revision 29
MDQ0067870253, Check Valves Low Flow EECW System, Revision 12
MDQ0999910034, NRC Generic Letter 89-10 Motor Operated Valve Evaluation, Revision 19
MDQ107420020078, MOV 1-FCV-74-0059 & MOV 1-FCV-74-0073, Operator Requirements and Capabilities, Revision 2
MDQ2073910103, MOV 2-FCV-073-0044, Operator Requirements and Capabilities, Revision 10
OTC-252, Thrust & Torque Calculations Tennessee Valley Authority Browns Ferry Nuclear Plant TVA Ref. No. 87NNQ-75105A S.O. #00632, Revision 2
OTC-258, Thrust & Torque Calculations Tennessee Valley Authority Browns Ferry Nuclear Plant TVA Ref. No. 87NNQ-75105A S.O. #00632, Revision 0
Zachry Calculation 15-115, Browns Ferry Nuclear Plant U2 "2A" and "2C" Residual Heat Removal Heat Exchanger Thermal Performance Test Report (January 8, 2015), Revision 0
Zachry Calculation 12-025, Browns Ferry Nuclear Plant U3 "3A" and "3C" Residual Heat Removal Heat Exchanger Thermal Performance Test Report, Revision 0
MDQ0023980143, RHR Heat Exchanger Tube Plugging Analysis for Power Uprates, Revision 3
Uncontrolled MathCad Analysis of Heat Exchanger 2C Test Results
MDQ002320100019, RHRSW System Hydraulic Analysis for Units 1, 2, & 3 RHR Heat Exchangers, Revision 2
MDQ002320100019, RHRSW System Hydraulic Analysis for Units 1, 2, &3 RHR Hx, Revision 2,
MDQ0000732012000033, HPCI In-Service Testing Instrument/Measurement Uncertainty Evaluation, Revision 1
MDQ099920040040, HPCI & RCIC Test Requirements, Revision 11
EDQ2000-870548, 4KV SHDN BD ‘A’ Protective Devices Rating/Setting Based on Analysis, Revision 44
EDQ2000870046, Load Study Diesel Generator Batteries, Revision 22
EDQ0057-2002-0022, Board Short Circuit Summary, Attachment 23, Sheet No 14 &16, Revision 14
EDQ008220090008, Evaluation of 59N Overvoltage Relays for 4.16KV Standby Generator output Breaker, Revision 0
ED-Q0248-870045, DC Short Circuit Calculation, Revision 013
EDQ024820030002, 4160 Shutdown Boards A, B, C, D and 3EB 250V DC Battery Load Study, Voltage Drop, and Short Circuit Calculation, Revision 15
EDQ005720020022, 4,16KV and 480V Busload, Voltage Drop and Short Circuit Calculation, Revision 015
EDQ0057920034, 4,16KV and 480V Busload, Voltage Drop and Short Circuit Calculation, Revision 097
EDQ0-248-2002-0042, 250V DC Unit Batt. Load Study, VD, SC, and Batt Capacity for LOCA/LOOP, Station Blackout and Appendix R Analysis for Unit/Shutdown Board Battery, Revision 069
EDQ199920020061, 250V DC Bus and Cable Protection and Breaker/Fuse Coordination, Revision 028
EDQ0248870041, 250V DC Unit Battery Load Study, Revision 030
EDQ199920020071, 480VAC Motor Control Centers, Cable and Bus Protection / Breaker Coordination, Revision 028
EDQ199920020061, 250V DC Bus and Cable Protection and Breaker/Fuse Coordination, Revision 28
EDQ2000870550, Cable and Bus Protection/ Breaker/ Fuse Coordination for 250V DC System, Revision 042
MD-Q3031-880242, Battery Rooms No.1 and No. 2 Ventilation Requirements, Revision 006

Design Basis Documents

BFN-50-7067, Emergency Equipment Cooling Water System, Revision 22
BFN-50-7073, High Pressure Coolant Injection System, Revision 29
BFN-50-7074, Residual Heat Removal System, Revision 28
BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing), Revision 5
BFN-50-C-7106, Equipment Seismic / Structural Qualification (ESQ), Revision 5
BFN-50-7003, General Design Criteria document, Reactor Feedwater System, Revision 17
BFN-50-7074, General Design Criteria Document, Residual Heat Removal Document, Revision 28
BFN-50-7030, General Design Criteria Document, Diesel Generator Building Environmental Control System, Revision 9
General Design Criteria Document N. BFN-50-7082, Standby Diesel Generator, Revision 24
BFN-50-7200C, Design Criteria, 250V DC Power Distribution System, Revision 07
BFN-50-7200D, Design Criteria, 480V AC Auxiliary Power System, Revision 11

Condition Reports (CRs)

162116	631889	604811
372659	601332	943519
376897	585893	941033
458431	502085	448016
767120	486298	1038747
840145	450502	1039036
859118	418254	682254
883568	403005	676678
803616	399998	696876
1053810	365489	818032
1101063	363213	824238
1073295	315818	958491
1067077	674040	1086479
1062464	376897	750154
1001458	359841	726606
959794	359804	732511
922629	374121	981859
789476	940890	703242
724825	940915	
682544	855648	

Work Orders

110966305	114307935	114392094
113183170	114307946	114399374
113473867	114307965	114422200
113680271	114307978	114433491
113804515	114307985	114486349
113805143	114307991	114489544
113807084	114308011	114511426
113862450	114308020	114522878
114057716	114308026	114594844
114255494	114324344	
114307920	114324345	
114307928	114391505	

114630300
114696989
114760056
114760107
114787126
114787138
114787163
114787255
114983194
115123308
115123309
115123311
115123313
115190805
115190828
115190839
115190862
115190879
115262326
115262333
115278857
115278867
115302871
115336096
115363391
115376833
115378083
115541064
115555503
115555508
115555515
115555535
115555541
115623747
115623749
115623753
115689637
115698562
115698588
115698626
115698637
115698650
115700098
115748415
115754141
115755978
115755983
115755985
115878821
115879055
115882857
115945434
115967782
116151248
116151266
116151280
116159644
116207810
116325637
116325690
116331648
116885100
116885120
117004985
117120184
117169291
117332970
114724730
114724708
115807121
114724713
113624211
114120807
116234079
115752804
110931890
115204443
115763456
115763454
115763453
09-724159-000
113218814
114821753
115562305
115562306
116032032
116474759
116474770

<u>Miscellaneous</u>

0-TI-360, Containment Leak Rate Programs, Revision 40
0-TI-362, Inservice Testing Program, Revision 48
BFN Magnesium Rotors White Paper
BFN-RAH-306, Valve Qualification, Revision 2
BFN-VTD-C665-0050, Crane List 900 and List 150 Gate Valves, Revision 9
BFN-VTD-D012-0020, Daniel Flow Products Chexter Check Valves, Revision 2
BFN-VTD-L200-0220, Limitorque Type SMB 10 CFR Part 21 Notifications Maintenance Updates and Technical Updates, Revision 17
BFN-VTD-W030-0030, Walworth Motor-Operated Valves, Revision 19
Browns Ferry Nuclear Plant Units 1, 2, and 3- Issuance of Amendments Regarding Deletion of Containment Isolation Valve Local Leak Rate Testing (TAC Nos. ME1801, ME1802, and ME1803), dated 3/22/10
BWROG-TP-09-005, Inspection of Motor Operated Valve Limitorque AC Motors with Magnesium Rotors, Revision 0
Crane Certificate of Compliance for PO 193440-2, dated 3/11/11
DCN 69896, Perform JOG Updates for Valve 2-FCV-73-44, Revision A
DCN 69898, Perform JOG Updates for Valve 1-FCV-74-60, & 1-MVOP-74-57 & 71, RHR System (074), Revision A
DS-M18.2.21, Motor Operated Valve Thrust and Torque Calculations, Revision 21
DS-M18.2.22, MOV Design Basis and JOG Review Methodologies, Revision 6EQV 71422, Revise DC MOV Calculations to Address PERs 711171, 740110, 765395, 847070, and 859627. Modify MOVs as Required to Meet Stroke Time Requirements, Revision A
EQV 71553, Revise MOV Calculation to Improve Margin and Update Drawings, Revision A
ER-5.0, Equipment Inaccuracy Summary for Motor Operated Valves, Revision 27
Motor Operated Valves Component Health Report, dated 1/1/15 – 6/30/15
MOV-WP-159, Commonwealth Edison White Paper 159: Capability Requirements for MOVs with Potential for Stem / Stem Nut or Worm / Worm Gear Locking, Revision 0
NPG-SPP-09.26.14, Motor Operated Valve Program, Revision 0
OPL171.042, High Pressure Coolant Injection (HPCI), Revision 21
OPL171.051, Emergency Equipment Cooling Water, Revision 17
PIC 70749, Incorporate Required Stem Thrust from EPRI PPM Evaluation, Revision A
QIR CEBBFN91021, Valve Qualification, Revision 0
QIR CEBBFN92017, Qualification Review for Additional Valve Thrust Values Identified by Generic Letter 89-10, Revision 0
QIR CEBBFN93007, Qualification Review for Additional Valve Thrust Values Identified by Generic Letter 89-10, Revision 0
QIR LMEBFN91035, Valve Qualification, Revision 0
QIR LMEBFN92020, Qualification Review for Additional Valve Thrust Values Identified by Generic Letter 89-10, Revision 0
System Health Report, HPCI, dated 6/1/15-9/30/15
System Health Report, RHR, dated 6/1/15-9/30/15
System Health Report, RHRSW/EECW, dated 6/1/15-9/30/15
Technical Specifications Change 465- Revision of Technical Specifications to Eliminate Unnecessary Water Local Leak Rate Tests, dated 7/27/09
TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan (NQAP) (Quality Assurance Program Description), Revision 31
0-TI-362(Bases), IST Program basis Document, Revision 10
0-TI-360, Containment Leak rate Program, Revision 40
Eddy Current Inspection Results

Residual Heat removal 2C, dated 1/27/2011
Past Operability evaluation Documentation for PER 1001458, dated 3/26/2015
Past Operability evaluation Documentation for PER 732555 and 741041, dated 7/2/2013
Past Operability evaluation Documentation for PER 750848 and 750858, dated 7/24/2013
Prompt Determination of Operability Documentation for PER 732555, dated 5/30/2013
Functional Evaluation for SR 314747, Revision 0
BFN-50-715, Environmental design, Revision 10
MathCad Analysis for the 2C RHR Heat Exchanger with 305 Tubes plugged by Shells, dated 3/25/2015
Calibration Certificate No. 45904 for Graftel instruments used in testing of the 2A and 2C RHR Heat Exchangers, dated 11/18/2014
Calibration Certificate No. 45904 for Graftel instruments used in testing of the 1A and 1C RHR Heat Exchangers, dated 11/11/2015
50.59 Review for Erection of Scaffolds near the RHR Heat Exchangers, dated 6/14/2007
ANSI/ASME PTC 6 Report, Guidance for Evaluation of Measurement Uncertainty in Performance Tests of Steam Turbines, 1985
Letter Zachry Nuclear Engineering to TVA – Watts Bar Nuclear Plant, titled Calculated Fouling Resistances in PROTO-HX, dated 1/15/2015
Past Operability Evaluation Documentation for PER 1001458, dated 3/26/2015
BFN-VTD-P160-0030, Instruction manual for PERFEX Residual Heat Removal Heat Exchangers, Revision 8
BFN-VTD-A585-0060, Instruction Manual Atwood and Morrill 24 Inch Check valve, Revision 3
Main Steam System Health Reports, dated 10/1/2014 – 1/31/2015
Main Steam System Health Reports, dated 2/1/2015 – 5/31/2015
Functionality Evaluation Documentation for CR 604811, Revision 2
TVA-BFN-TS-436, Increased MSIV Leakage Rate Limits Change Request
TVA-BFN-TS-399, TS Change Request Increased MSIV Leakage Rate and Exemption from 10 CFR Appendix J
50.59 Applicability Determination Guidance, Attachment 1
MSIV Leak Rate Tabulation Evaluation
Licensee Event Report 50-260/1993
ANSI/ANS-56.8-1994, Standard for Containment System Leakage Testing Requirements, dated 8/4/1994
General Electric SIL No. 329, Fire Related to MSIV Fluid Leak
RHRSW 3Q 2015 System Health Report
HPCI 3Q 2015 System Health Report
Main Steam 3Q 2015 System Health Report
DCN 70491, NFPA-805 SDBD Bus Lockout Relays, Crosstie Relays, Fault Relays, 1/9/12
GEH-2058L, GE Instructions Auxiliary Relays Type HEA61
BFN-VTM-P318-0010, Vendor Technical Manual for Emergency Diesel Units 1, 2, & 3, Revision45
BFN-VTD-SI06-0050, Installation, Operation and Maintenance Instructions for Siemens Type-3AFS Vacuum Circuit Breaker, Revision 0
IEEE 242 "Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems, 1986
NEMA MG 1, Motors and Generators, Revision 1 2007
NRC IN 90-43: Mechanical Interference with Thermal Trip Function in GE Molded-Case Circuit Breakers,
IEEE 450-1987, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications

BFN-VTM-G080-1430, Vendor Manual, General Electric 480-V Reactor MOV Boards, Revision 24
BFN-VTD-C173-0090, Vendor Manual, Specification Sheet C&D Batteries KCR-Lead Calcium Batteries, Revision 01
BFN-VTD-C173-0020, Vendor Manual, Standby Battery Vented Cell Installation & Operating Instructions, Revision 05
BFN-VTD-EN17-0010, Enersys Power Safe Performance Specifications, Revision 0
EPI -0-248-BAT005, Annual Inspection of 250V DC Main Battery Banks 1, 2, 3 and Associated Chargers, Revision 0021
ETAP 7.0N, Electrical Calculation, Load Flow set up Shutdown on USST 1B, 3B, dated 08-14-2104 study case: 500-N2B3B
EPI -0-281-RLY001, Relay Calibration and Functional Tests on 250V DC Reactor MOV Boards, Revision 0013
OPL171.036, AC Power Distribution, Revision 13, Revision 0016
OPL171.037, DC Systems, Revision 13, Revision 0013
FSAR BFN section 8.4 Normal Auxiliary Power System
FSAR BFN section 8.6, 250-V DC Power Supply and Distribution
BFN-Unit 2, Technical Specification Bases, section B.3.8 Electrical Power Systems, Revision 52
ECI-0-000-BKR008, Testing and Troubleshooting of Molded Case Circuit Breakers and Motor Starter Overload Relays, Revision 0105
IN 2013-05, Battery Expected Life and its Potential Impact on Surveillance requirements, 03/19/13
0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System, Revision 0147
0-GOI-300-1/ATT-11, Attachment 11, Control Bay Operator Round Log, Revision 0223
ECI-0-000-BKR008, Preventative Maintenance, Testing and Troubleshooting of Molded Case Circuit Breakers and Motor Starter Overload Relays, dated 10/25/10, Revision 0105
System Health Report (02/1/2013 – 5/31/2013), DC Electrical, Unit 0
System Health Report (10/1/2013 – 1/31/2014), DC Electrical, Unit 0
System Health Report (06/1/2014 – 9/30/2014), DC Electrical, Unit 0
System Health Report (10/1/2014 – 1/31/2015), DC Electrical, Unit 0
System Health Report (02/1/2015 – 5/31/2015), DC Electrical, Unit 0
System Health Report (02/1/2013 – 5/31/2013), 480V AC Power Distribution, Unit 0
System Health Report (02/1/2014 – 5/31/2015), 480V AC Power Distribution, Unit 0
System Health Report (10/1/2014 – 1/31/2015), 480V AC Power Distribution, Unit 0
System Health Report (02/1/2015 – 5/31/2015), 480V AC Power Distribution, Unit 0
System Health Report (06/1/2015 – 9/30/2015), 480V AC Power Distribution, Unit 0
DCN 51090, BFNP Unit 1 Recovery-Electrical Lead DCN-System 57-4-480V Electrical Distribution (CB Bldg.), closed date 11/06/13
DCN 70747, GE AK Low Voltage Circuit Breaker Replacement, dated 6/25/15
PIC, 70192, Revise ICRDS and DCA's, Revision A, closed 6/7/13
DCN W-17274A, Replace 250V DC Unit Bat's 2 & 3, verified date 2/11/92
EQV 70294, Replace Magnesium rotors in the Drywell, 01/02/14
B.3.8.4, Technical Specification, DC Sources – Operating, Revision 0
B.3.8.5, Technical Specification, DC Sources – Shutdown, Revision 0
B.3.8.6, Technical Specification, Battery Cell Parameters, Revision 0
B.3.8.7, Technical Specification, Distribution Systems – Operating, Revision0

Corrective Action Documents Written Due to this Inspection

1117106
1117381
1114779
1115509
1112661
1117132
1117111
1117078
1116214
1113238
1111001
1110468
1111562
1111309
1111302
1111007
1110980
1110968
1110961
1110950
1110938
1110930
1110927
1110914
1110687
1110218