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

May 13, 2016

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2016001, 05000260/2016001, AND 05000296/2016001

Dear Mr. Shea:

On March 31, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. On April 19, 2016, the NRC inspectors discussed the results of this inspection with Mr. S. Bono and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented 8 findings which were determined to be of very low safety significance (Green) in this report. Seven of these findings involved violations of NRC requirements. The NRC is treating these violations as noncited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violation 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, D.C. 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the 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 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’s Public Document Room or from the Publicly Available Records (PARS) component of NRC’s Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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:
IR 05000259/2016001, 05000260/2016001
and 05000296/2016001
w/Attachment: Supplemental Information

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Letter to Joseph W. Shea from Alan Blamey dated May 13, 2016.

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2016001, 05000260/2016001, AND 05000296/2016001

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2016001, 05000260/2016001, 05000296/2016001
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:	January 1, 2016, through March 31, 2016
Inspectors:	D. Dumbacher, Senior Resident Inspector T. Stephen, Resident Inspector A. Ruh, Resident Inspector A. Nielsen, Senior Health Physics Inspector R. Kellner, Senior Health Physics Inspector R. Williams, Senior Reactor Inspector S. Sanchez, Senior Reactor Inspector C. Fontana, Senior Reactor Inspector E. Powell, Reactor Inspector
Approved by:	Alan Blamey, Chief Reactor Projects Branch 6

Enclosure

TABLE OF CONTENTS

Other Activities

SUMMARY

IR 05000259/2016001, 05000260/2016001, 05000296/2016001; 01/01/2016–03/31/2016; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Operability Determinations and Functionality Assessment, Surveillance Testing, Occupational Radiation Protection, Drill Evaluation, and Follow-up of Events and Notices of Enforcement Discretion.

The report covered a three month period of inspection by resident and regional inspectors. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using 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.

NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing Non-Cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, Design Control was identified for the licensee's failure to properly install the Unit 2 High Pressure Coolant Injection (HPCI) turbine steam admission valve packing assembly. The licensee installed a valve packing type that was not as specified in design control drawings and due to inadequate maintenance drawings installed the packing gland follower upside down. Upon discovery of the packing failure, the licensee took action to isolate the associated steam leak and declare the HPCI system inoperable. Repairs were completed and tested on September 19, 2015. The licensee entered the issue into their corrective action program as CRs 1114188 and 1127172.

  The performance deficiency was more-than-minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the failure to maintain the design features led to the loss of operability of the HPCI system when valve 2-FCV-73-16 packing failed and HPCI was isolated to stop the steam leak. This finding was evaluated in accordance with NRC IMC 0609, Appendix A, Exhibit 2 "Mitigating Systems Screening Questions," dated June 19, 2012. The finding was screened to Green because HPCI would have been able to perform its design basis function with the steam leak. The inspectors determined that the finding had a cross cutting aspect of Design Margins because the licensee allowed non-equivalent packing material to be installed in the Unit 2 HPCI steam admission valve. (H.6) (1R15)

- Green. A self-revealing Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions, was identified for the licensee's failure to take corrective action following the discovery of a significant steam leak from the packing gland of the Unit 2 HPCI steam inlet isolation valve, 2-FCV-73-16. Specifically, the licensee failed to correctly classify the severity of the leak on 2-FCV-73-16 as described in NPG-SPP-06.8, Leak Reduction Program, and allowed the condition to degrade until packing failure. Upon discovery of the packing failure, the licensee took action to isolate the associated steam leak and declare the HPCI system inoperable. Repairs were completed and tested on September 19, 2015. The licensee entered the issue into their corrective action program as CR 1082405.

  The performance deficiency was determined to be more-than-minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, misclassification of the leak severity as minor led to the loss of function of the HPCI system when valve 2-FCV-73-16 packing degraded until failure and HPCI was isolated to stop the steam leak. This finding was evaluated in accordance with NRC IMC 0609, Appendix A, Exhibit 2 "Mitigating Systems Screening Questions," dated June 19, 2012. The finding was screened to Green because HPCI would have been able to perform its design basis function with the steam leak. The inspectors determined that the finding had a cross cutting aspect of Resolution because the licensee did not take timely corrective action to repair the Unit 2 HPCI steam leak before it lead to a Safety System Functional Failure. (P.3) (1R15)

- Green. An NRC identified finding (FIN) for failure to meet TVA procedure NETP-116.3, "Inservice Testing Program Preconditioning Guidelines," because unacceptable preconditioning of the Unit 2 Reactor Core Isolation Cooling (RCIC) steam supply valve occurred prior to quarterly In-Service Test (IST). Specifically, the preconditioning was unacceptable because the testing sequence was avoidable, it masked the actual as-found condition of the valve, and it could possibly result in an inability to verify the operability of the valve. As an immediate corrective action, the licensee performed an evaluation that determined the valve remained operable. The finding was entered into the licensee's corrective action program as CR 1159463 .

  The performance deficiency was more-than-minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Additionally, if left uncorrected, the performance deficiency could lead to a more significant safety concern. Specifically, the licensee's justification of this particular preconditioning event could be applied to justify additional, avoidable, preconditioning events and possibly result in an inability to verify the operability of components. This finding was evaluated in accordance with NRC IMC 0609, Appendix A, Exhibit 2 "Mitigating Systems Screening Questions," dated June 19, 2012. The inspectors determined the finding was Green because the finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its TS allowable outage time, did not result in a loss of function of non-TS equipment, and did not involve the loss of equipment or function specifically designed to mitigate an external event. The inspectors determined that the finding had a cross-cutting aspect in the Human Performance area of Consistent Process [H.13], because individuals did not complete the required preconditioning evaluation forms described in licensee procedure NETP-116.3, which would have challenged the validity of the licensee's original determination of acceptability. (1R22)

- Green. An NRC identified non-cited violation (NCV) of Technical Specification (TS) 5.4.1, Procedures, for the licensee's failure to implement OPDP-8, Operability Determinations and LCO Tracking. Specifically, the licensee failed to track the applicability of condition 'A' of TS LCO 3.6.1.3 upon discovery of the equipment failure related to the Residual Heat Removal (RHR) Shutdown Cooling (SDC) inboard suction valve as described in LER 05000296/2014-003-00. As an immediate corrective action, the licensee entered the violation into the corrective action program as CR 1115172.

  The performance deficiency was more-than-minor because, if left uncorrected, would have the potential to lead to a more significant safety concern. Specifically, this failure was indicative of a programmatic weakness with the licensee's evaluation of certain logic circuit failures which can result in misapplication of the allowances of TS LCO 3.0.6 and inappropriate TS LCO entries. The inspectors determined that this type of error was likely to recur which could lead to worse errors if uncorrected. The inspectors determined the finding was Green because the error did not result in an actual open pathway in the physical integrity of reactor containment, containment isolation system or heat removal components. The inspectors determined that the finding had a cross-cutting aspect of Training in the area of Human Performance because the finding was indicative of a knowledge gap among the operations department (H.9). (4OA3)

Cornerstone Occupational Radiation Safety

Green. A self-revealing, Non-cited Violation (NCV) of Technical Specification (TS) 5.7.1, was identified for a worker who entered a High Radiation Area (HRA) without proper authorization. Specifically, the worker entered a posted HRA located outside the Radwaste Ventilation Equipment Room without receiving a HRA briefing, and subsequently received a dose rate alarm. This issue was entered into the licensee's corrective action program as Condition Report (CR) 1072342, and the licensee took immediate corrective actions including surveys of the area, and restricting the worker's access to the Radiologically Controlled Area.

The performance deficiency was greater than minor because it was associated with the Occupational Radiation Safety cornerstone attribute of Program and Process (Monitoring and Radiation Protection (RP) Controls) and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The inspectors determined the finding to be of very low safety significance (Green) because it was not related to As Low As Reasonably Achievable (ALARA) planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. This finding involved the cross-cutting aspect of Human Performance, Procedural Adherence [H.8] because the event was a direct result of the worker's failure to adhere to requirements for HRA access. (2RS1)

Green. A self-revealing, NCV of 10 CFR 20.1902(b), with two examples, was identified for the failure to post multiple HRAs. Specifically, areas within the Unit 2 (U2) Control Rod Drive Rebuild Room and U2 Reactor Water Cleanup Holding Pump Room contained dose rates exceeding 100 mrem/hr at 30 cm and remained unposted for several months during 2015. These issues were entered into the licensee's corrective action program as CR 1017294, CR 1023385, and CR 1119944, and the licensee took immediate corrective actions to correctly post the areas, performed surveys to evaluate the extent of condition, and performed an Apparent Cause Evaluation.

The performance deficiency was greater than minor because it was associated with the Occupational Radiation Safety cornerstone attribute of Program and Process (Monitoring and RP Controls) and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The inspectors determined the finding to be of very low safety significance (Green) because it was not related to As Low As Reasonably Achievable (ALARA) planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. This finding involved the cross-cutting aspect of Human Performance, Documentation [H.7] because the unposted high radiation areas were a direct result of the failure to identify documented radiological conditions that required additional posting and control. (2RS1)

Cornerstone: Public Radiation Safety (RS)

Green. The inspectors identified a NCV of 10 CFR 71.5 for the failure to include the correct Proper Shipping Name (PSN) on radioactive material shipping papers in accordance with the requirements of Department of Transportation (DOT) regulation 49 CFR 172.202. This resulted in multiple Low Specific Activity (LSA) shipments containing quantities exceeding an A_2 value being shipped as "UN2915, Radioactive Material, Type A Package". The licensee documented this issue in CR 1145617 and took immediate corrective actions including updating the software used to perform shipping activities and additional training of personnel.

The performance deficiency was greater than minor because it was associated with the Public Radiation Safety Cornerstone, Program & Process attribute (transportation program), and adversely affected the associated cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The inspectors determined the finding to be of very low safety significance (Green) because the issue involved transportation, but there were no radiation limits exceeded, and there was no package breach. In addition, it did not involve a Certificate of Compliance or low-level burial problem, nor was there a failure to make notifications or provide emergency response information. The finding has a cross-cutting aspect in the area of Human Performance, Training [H.9], because the DOT requirements pertaining to LSA shipments were not well understood. (2RS8)

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50.54(q)(2), for the licensee's failure to maintain the effectiveness of its emergency plan by ensuring procedures for use by the emergency response organization are maintained and up-to-date as required by 10 CFR 50.47(b)(16). Corrective actions already taken were implementation of a revision (49) to EPIP-5, effective January 7, 2016, essentially replacing Section 3.6 and references to appropriate Appendices, and a broader scope EOC to review all site EPIPs to ensure no other inadvertent omissions were made.

  The inspectors determined that the performance deficiency was more than minor because it was associated with the procedure quality attribute of the Emergency Preparedness (EP) cornerstone, adversely affected the associated cornerstone objective, and may have been used had an emergency been declared. The finding was evaluated using the EP significance determination process and was identified as having very low safety significance (Green) because it was a failure to comply with NRC requirements and was not a loss of the planning standard function. The finding was associated with a cross-cutting aspect in the Evaluation component of the Problem Identification and Resolution area because the licensee failed to thoroughly evaluate a similar issue at one of its other sites to ensure extent of conditions commensurate with their safety significance are thoroughly resolved. [P.2] (Section 1EP6.2)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at 100 percent of rated thermal power (RTP) except for one unplanned and two planned downpowers. The unplanned downpower to 13 percent of RTP on March 25, 2016 was to perform a drywell entry to refill an oil reservoir on the 1B recirculation pump. The two planned downpowers for maintenance occurred on February 9, 2016 and February 13, 2016.

Unit 2 operated at 100 percent of RTP except for two planned downpowers for maintenance on February 12, 2016 and February 19, 2016.

Unit 3 continued its coastdown until the planned refueling outage that began on February 20, 2016. The unit was restarted on March 26, 2016 and achieved 100 percent of RTP on March 31, 2016.

REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors reviewed the licensee's preparations to protect risk-significant systems from a tornado watch on February 2, 2016. The inspectors evaluated the licensee's implementation of adverse weather preparation procedures and compensatory measures, including operator staffing, before the onset of and during the adverse weather conditions. The inspectors reviewed the licensee's plans to address the short and long term effects that may result from a tornado event. The inspectors verified that operator actions specified in the licensee's adverse weather procedure maintain readiness of essential systems. The inspectors verified that required surveillances were current, and completed before the onset of anticipated adverse weather conditions. The inspectors also verified that the licensee implemented periodic equipment walkdowns or other measures to ensure that the condition of plant equipment met operability requirements. Documents reviewed are listed in the attachment. This constituted one Impending Adverse Weather sample as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

.2 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

After the licensee completed preparations for seasonal low temperature, the inspectors walked down the Emergency Diesel Generators, Intake Structure, and the Service Water

Pump Rooms. These systems were selected because their safety related functions could be affected by adverse weather. The inspectors reviewed documents listed in the Attachment, observed plant conditions, and evaluated those conditions using criteria documented in the Inspection Procedure. Documents reviewed are listed in the attachment. This activity constituted one Readiness for Seasonal Extreme Weather conditions inspection sample as defined in IP 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors conducted partial equipment alignment walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, while the other subsystems were inoperable or out of service. The inspectors reviewed the functional systems descriptions, Updated Final Safety Analysis Report (UFSAR), system operating procedures, and Technical Specifications (TS) to determine correct system lineups for the current plant conditions. The inspectors performed walkdowns of the systems to verify that critical components were properly aligned and to identify any discrepancies which could affect operability of the redundant train or backup system. Documents reviewed are listed in the attachment. This activity constituted four Equipment Alignment Partial Walkdown inspection samples, as defined in Inspection Procedure 71111.04.

- Unit 3 Primary Containment with a focus on rising oxygen concentrations
- Unit 1 RCIC while HPCI was out of service
- Unit 3 Spent Fuel Pool following a full core offload during a refueling outage
- Unit 3 Torus with a focus on structural integrity and coating reviews.

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 2 Reactor Core Isolation Cooling system.

The inspectors reviewed relevant portions of the Updated Final Safety Analysis Report (UFSAR) and TS. This detailed walkdown also verified electrical power alignment, the condition of applicable system instrumentation and controls, component labeling, pipe hangers and support installation, and associated support systems status. The inspectors examined applicable System Health Reports, open Work Orders (WOs), and any previous Condition Reports (CRs) that could affect system alignment and operability. Documents reviewed are listed in the attachment. This activity constituted one Equipment Alignment Complete Walkdown inspection sample, as defined in Inspection Procedure 71111.04.

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 the fire areas (FA) and fire zones (FZ) listed below. Selected FAs/FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. 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 Report, Volumes 1 and 2, including the applicable Fire Hazards Analysis, and Pre-Fire Plan drawings, to verify that the necessary firefighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, was in place. Documents reviewed are listed in the attachment. This activity constituted five Fire Protection Walkdown inspection samples, as defined in Inspection Procedure 71111.05.

- Fire Area 2-2, Unit 2 Reactor Building, Elevation 519' to 565', from column line R14 to 10' west of column line R11
- Fire Area 25-1, Intake Pumping Station
- Fire Area 9, Unit 2 Reactor Building, Elevation 621', Electrical Board Room 2A and 250v Battery Room
- Fire Area 10, Unit 2 Reactor Building, Elevation 621', 480v Shutdown Board Room 2A
- Fire Area 11, Unit 2 Reactor Building, Elevation 621', 480v Shutdown Board Room 2B

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

.1 Annual Review of Cables Located in Underground Bunkers/Manholes

a. Inspection Scope

The inspectors conducted a review of licensee inspections of safety-related cables located in underground bunkers/manholes subject to flooding. Specifically, inspectors reviewed maintenance records and observed an inspection of hand hole 15 and hand hole 26 to determine if water was present and, if found, whether it would affect safety-related system operation. In addition, the inspectors reviewed the licensee's CAP to ensure that the licensee was identifying underground cabling issues and that they were properly addressed for resolution. Documents reviewed are listed in the Attachment. This activity constituted one underground cable inspection sample, as defined in Inspection Procedure 71111.06.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Review of Heat Exchanger Performance

a. Inspection Scope

The inspectors observed the thermal performance test of the Unit 1 B and D RHR heat exchangers to determine whether there were any previously undetected adverse performance trends, whether the acceptance criteria and results appropriately considered differences between testing conditions and design conditions; and whether test results were appropriately categorized against pre-established acceptance criteria. The inspectors also reviewed work documents detailing observations and results of the last internal inspection of the heat exchangers. Documents reviewed are listed in the Attachment. The inspectors completed one heat sink performance inspection sample as defined in IP 71111.07.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

Non-Destructive Examination Activities and Welding Activities

From March 7–11, 2016, inspectors conducted an onsite review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the reactor coolant system boundary, risk-significant piping and component boundaries, and containment boundaries in Unit 3.

The inspectors either directly observed or reviewed the following non-destructive examinations (NDEs) mandated by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code of Record: 2007 Edition with 2008 Addenda) to evaluate compliance with the ASME Code, Section XI and Section V requirements, and if any indications or defects were detected, to evaluate if they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement. The inspectors also reviewed the qualifications of the NDE technicians performing the examinations to determine whether they were current, and in compliance with the ASME Code requirements.

- Ultrasonic Examination (UT) of DSRHR-3-01, elbow to pipe weld, Class 1 (observed)
- UT of DSRHS-3-08, tee to pipe weld, Class 1 (observed)
- Liquid Penetrant Examination (PT) of EECW-3-011-113 , valve to pipe weld, Class 3 (reviewed)
- Magnetic Particle Examination (MT) of weld HPCI-3-026-001A, integral welded attachment, Class 2 (reviewed)
- MT of component 3-47B455-631-IA, integral welded attachment, Class 2 (reviewed)
- Visual Examination (VT-3) of component 3-47B455-631-IA, integral welded attachment, Class 2 (reviewed)

The inspectors reviewed the following welding activities, qualification records, and associated documents in order to evaluate compliance with procedures and the ASME Code, Section XI and Section IX requirements. Specifically, the inspectors reviewed the work order (WO), repair and replacement plan, weld data sheets, welding procedures, procedure qualification records, welder performance qualification records, and NDE reports.

- WO 115569837, Removal of an indication found an integral welded attachment, Class 2
- WO 116930078, Replacement of Header Supply Valve to SW RHR Room, Class 3

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

<u>Identification and Resolution of Problems</u>

The inspectors reviewed a sample of ISI-related issues entered into the corrective action program to determine if the licensee had appropriately described the scope of the problem, and had initiated corrective actions. The review also included the licensee’s consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The inspectors completed one in-service inspection sample as defined in IP 71111.08.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification and Performance (71111.11)

.1 Licensed Operator Requalification

a. Inspection Scope

On January 4, 2016, the inspectors observed a licensed operator training session for an operating crew according to the Unit 2 Browns Ferry Training Plan OPL 175S.039 Revision 0.

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

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

The inspectors assessed the licensee's ability to assess the performance of their licensed operators. The inspectors reviewed the post-examination critique performed by the licensee evaluators, and verified that licensee-identified issues were comparable to issues identified by the inspector. The inspectors reviewed simulator physical fidelity (i.e., the degree of similarity between the simulator and the reference plant control room, such as physical location of panels, equipment, instruments, controls, labels, and related form and function). Documents reviewed are listed in the attachment. 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)

.1 Routine

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 in accordance with 10 CFR 50.65(b) of the MR; (4) Characterizing reliability issues for performance monitoring; (5) Tracking unavailability for performance monitoring; (6) Balancing reliability and unavailability; (7) Trending key parameters for condition monitoring; (8) System classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); (9) Appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2); and (10) Appropriateness and adequacy of 10 CFR 50.65 (a)(1) goals, monitoring and corrective actions. The inspectors compared the licensee's performance against site procedures. The inspectors reviewed, as applicable, work orders, surveillance records, PERs, system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met.

Documents reviewed are listed in the attachment. This activity constituted two Maintenance Effectiveness inspection samples, as defined in Inspection Procedure 71111.12.

- Unit 1 RHR system
- A3 and D2 Residual Heat Removal Service Water (RHRSW) pump adverse vibration trends driving increased testing frequency.

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) and/or using 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. Documents reviewed are listed in the attachment. This activity constituted five Maintenance Risk Assessment inspection samples, as defined in Inspection Procedure 71111.13.

- Unit 2 HPCI outage on January 5, 2016.
- Unit 3 loss of main steam bypass valves to the main condenser
- Unit 3 3D 4kV shutdown board inoperable
- Unit 1 HPCI outage on January 28, 2016
- Unit 3 Yellow risk during the drain down of reactor vessel level to support reactor vessel flange repairs

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 PERs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the attachment. This activity constituted seven regular Operability Evaluation inspection samples, as defined in Inspection Procedure 71111.15.

- Normal Feeder breaker trip during C Emergency Diesel Generator (EDG) post maintenance testing (CR 1125878)
- Unit 2 Suppression Chamber Standby Gas Inboard Isolation Valve inoperable (CR 1137399)
- Revised Prompt Determination of Operability (PDO) for Unit 2 HPCI Steam Admission Valve (CRs 1127169, 1127172, and 1127173)
- 3A RHR pump handswitch failed (CR 1126697)
- HPCI discharge pipe in steam vault compartment with elevated temperature (CR 1121668)
- Low EECW flow from south header to D Diesel Generator (CR 1145025)
- Impact to 3C Diesel Generator from possible loss of load shed capability of Drywell Blower (CR 1127554)

b. Findings

.1 Failure to Maintain the Design Packing Features of the Unit 2 HPCI Turbine Steam Admission Valve

Introduction: A self-revealing Green Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, Design Control was identified for the licensee's failure to properly install the Unit 2 HPCI turbine steam admission valve packing assembly. The licensee installed a valve packing type that was not as specified in design control drawings and due to inadequate maintenance drawings installed the packing gland follower upside down. These issues led to a degrading packing leak in June and eventual failure in September 2015.

Description: The HPCI steam isolation valve 2-FCV-73-16 had been replaced as part of Design Change Number (DCN) 70578 in April 2013 with a new 10 x 8 inch Flex Wedge disc gate design. The licensee installed a live-loaded graphite packing system as part of the DCN. On June 19, 2015, the licensee documented that the Unit 2 HPCI steam admission valve, 2-FCV-73-16 had a packing leak. A work order was initiated and scheduled for December 14, 2015 to repair the steam leak. On July 16, 2015 NRC inspectors notified the licensee staff that the leak had worsened and was very loud. Again, on July 31st, NRC inspectors noted the steam leak was excessively loud and provided the licensee staff a video of the leak. The licensee re-inspected the valve and concluded that the leak was a packing leak of minor significance and that the component and system were operable. No engineering reviews were performed.

On September 16, 2015, valve stroke surveillance 2-SR-3.6.1.3 cycled valve 2-FCV-73-16. Approximately 13 minutes later, Operations personnel received a fire alarm and reports of significant steam in the Unit 2 HPCI room. The steam leak had actuated temperature sensors in the Unit 2 HPCI room designed to initiate fire suppression water and, for large steam leaks, to isolate the steam supply to the HPCI turbine. Quick action by the operators to manually isolate the turbine lessened the amount of steam entering the room. The leak rendered the HPCI pump inoperable. The licensee made an 8-hour notification (Event Notice 51398) per 10 CFR 50.72(b)(3)(v)(D) for a loss of HPCI system safety function. Following the steam leak on September 16th the licensee identified that the valve's gland follower had been installed upside down. After the failure the licensee re-reviewed the DCN package. The DCN issued valve detail drawing CD05897, which specified a different packing material than installed by the licensee. The installed packing with high Teflon content was verified by the licensee to be susceptible to an observed accelerated failure rate in the presence of a steam leak. Although the drawing specification stated "OR EQL", a formal equivalency evaluation was not performed by Engineering for the different packing material. An evaluation should have identified the concerns about the observed failure mechanism. The licensee Design Engineering staff determined that the installed packing (which contains Teflon) did not conform to the current design. The packing and gland follower were replaced and the HPCI turbine and steam admission valve re-tested successfully on September 19, 2015. The licensee initiated corrective actions to replace the packing on the steam admission valve for each of the three units. The licensee provided additional information following the issuance of the AV that allowed final significance determination.

<u>Analysis</u>: The inspectors determined that the failure to properly install the new HPCI turbine steam admission valve packing assembly per valve detail drawing CD05897 was a performance deficiency. Specifically, the licensee installed improper packing and installed the packing gland follower upside down. The performance deficiency was more-than-minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the failure to maintain design control led to the loss of operability of the HPCI system when valve 2-FCV-73-16 packing failed and HPCI was isolated to stop the steam leak. This finding was evaluated in accordance with NRC IMC 0609, Appendix A, Exhibit 2 "Mitigating Systems Screening Questions," dated June 19, 2012. The finding was screened to Green because HPCI maintained the ability to perform its design basis function in the degraded condition. The inspectors determined that the finding had a cross cutting aspect of Design Margins because the licensee allowed non-equivalent packing material to be installed in the Unit 2 HPCI steam admission valve. (H.6)

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion III, Design Control states, in part, that measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components. Contrary to the above, from April 16, 2013 to September 16th, 2015, the licensee failed to provide adequate measures to control correct packing material and parts installation for the Unit 2 HPCI steam admission valve 2-FCV-73-16. The valve's unsuitable packing material and the gland follower being installed upside down led to a degrading packing leak starting in June 2015 and eventual failure in September 2015. Upon discovery of the packing failure, the licensee took action to isolate the steam leak and declare the HPCI system inoperable. Repairs were completed and tested on September 19, 2015. The licensee is

developing corrective actions to resolve the engineering design issues. The licensee entered the issue into their CAP as CRs 1114188 and 1127172. This NCV closes out Apparent Violation (AV) 05000260/2015004-05 from Browns Ferry Integrated Inspection Report Number 05000259,260,296/2015004. This violation is being treated as an NCV, consistent with section 2.3.2 of the Enforcement Policy. (NCV 05000260/2015004-05, Failure to Maintain The Design Packing Features of the Unit 2 HPCI Turbine Steam Admission Valve).

.2 Failure to Identify Significant Steam Leak on the Unit 2 HPCI Turbine Steam Admission Valve

Introduction: A self-revealing Green Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions, was identified for the licensees failure to take corrective actions following the discovery of a significant steam leak from the packing gland of the Unit 2 HPCI steam inlet isolation valve, 2-FCV-73-16. Specifically, the licensee failed to correctly classify the severity of the leak on 2-FCV-73-16 as described in NPG-SPP-06.8, Leak Reduction Program, and allowed the condition to degrade until packing failure.

Description: On June 19, 2015, the licensee documented that the Unit 2 HPCI steam admission valve, 2-FCV-73-16 HPCI had a packing leak. A work order was initiated and scheduled for December 14, 2015, to repair the steam leak. On July 16, 2015, NRC inspectors notified the licensee staff that the leak had worsened and was very loud. Again, on July 31, 2015, NRC inspectors noted the steam leak was excessively loud and provided the licensee staff a video of the leak. The licensee re-inspected the valve and concluded that the leak was a packing leak of minor significance and that the component and system were operable. The licensee scheduled repairs for December 14, 2015. On September 16, 2015, valve stroke surveillance 2-SR-3.6.1.3 cycled valve 2-FCV-73-16. Approximately 13 minutes later, Operations personnel received a fire alarm and reports of significant steam in the Unit 2 HPCI room. The steam leak had actuated one temperature sensor in the Unit 2 HPCI room designed to initiate fire suppression water and, for large steam leaks, to isolate the steam supply to the HPCI turbine. Quick action by the operators to manually isolate the turbine lessened the amount of steam entering the room. The isolation rendered the HPCI pump inoperable. The licensee made an 8-hour notification (Event Notice 51398) per 10 CFR 50.72(b)(3)(v)(D) for a loss of HPCI system safety function.

The inspectors identified that using NPG-SPP-06.8, Leak Reduction Program, the appropriate characterization of the packing leak was a Category 1, Severity level 5, the highest possible severity leak. This characterization should have been used to assign work priorities for repairing the valve as described in NPG-SPP-07.1.4 Work Management Prioritization – On Line. If properly characterized as at least a Priority 2 – Urgent, this condition would have required the repair of the leak to be scheduled at the earliest opportunity within T-3 work week schedule (i.e. within a maximum of 30 days). Following the steam leak on September 16, 2015, the licensee identified that the valve's packing had failed causing the steam leak. The licensee determined that the mischaracterization of the packing leak severity was a direct cause of not ensuring corrective action was taken in a timely manner to address the steam leak. The packing and gland follower were replaced and the system re-tested successfully on

September 19, 2015. The licensee initiated corrective actions in CR 1082405 to inspect similar valve design changes on Units 1 and 3 and training for engineers to better understand severity classifications for steam leaks. The licensee provided additional information following the issuance of the AV that allowed final significance determination.

Analysis: The inspectors determined that the licensee's failure to correctly classify the significance of the leak on the Unit 2 HPCI turbine steam admission valve, 2-FCV-73-16, packing was a performance deficiency. Specifically, the licensee classified the steam leak on 2-FCV-73-16 as minor which was not in accordance with the requirements of NPG-SPP-06.8, which would have assigned the most significant classification of steam leak, Category 1, Severity level 5. The performance deficiency was determined to be more-than-minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, misclassification of the leak severity as minor led to the loss of function of the HPCI system when valve 2-FCV-73-16 packing degraded until packing failure and HPCI was isolated to stop the steam leak. This finding was evaluated in accordance with NRC IMC 0609, Appendix A, Exhibit 2 "Mitigating Systems Screening Questions," dated June 19, 2012. The finding was screened to Green because HPCI maintained the ability to perform its design basis function in the degraded condition. The inspectors determined that the finding had a cross cutting aspect of Resolution because the licensee did not take timely corrective action to repair the Unit 2 HPCI steam leak before it led to a Safety System Functional Failure. (P.3)

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviation, defective material, and equipment and non-conformances are promptly identified and corrected. Contrary to the above, from July 16, 2015 to September 16, 2015, the licensee failed to promptly identify and correct a condition adverse to quality associated with the Unit 2 HPCI system. Specifically, on July 16 and 31, 2015, the licensee failed to correctly identify the severity of the packing leak on the Unit 2 HPCI steam admission valve, 2-FCV-73-16, per procedure NPG-SPP-06.8. This precluded the licensee from taking appropriate actions to correct the steam leak commensurate with its significance allowing the degradation and ultimate failure of the valve packing. Upon discovery of the packing failure, the licensee took action to isolate the steam leak and declare the HPCI system inoperable. Repairs were completed and tested on September 19, 2015. The licensee entered this issue into the CAP as CR 1082405. This NCV closes out AV 05000260/2015004-06 from Browns Ferry Integrated Inspection Report Number 05000259,260,296/2015004. This violation is being treated as an NCV, consistent with section 2.3.2 of the Enforcement Policy. (NCV 05000260/2015004-06, Failure to Identify Significant Steam Leak on the Unit 2 HPCI Turbine Steam Admission Valve).

1R18 Plant Modifications (71111.18)

.1 Permanent Plant Modifications

a. Inspection Scope

The inspectors verified that the plant modification(s) listed below did not affect the safety functions of important safety systems. The inspectors confirmed the modifications did not degrade the design bases, licensing bases, and performance capability of risk significant structures, systems and components. The inspectors also verified modifications performed during plant configurations involving increased risk did not place the plant in an unsafe condition. Additionally, the inspectors evaluated whether system operability and availability, configuration control, post-installation test activities, and changes to documents, such as drawings, procedures, and operator training materials, complied with licensee standards and NRC requirements. In addition, the inspectors reviewed a sample of related corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with modifications. Documents reviewed are listed in the attachment. This activity constituted two Plant Modification samples, as defined in Inspection Procedure 71111.18.

- DCN 71313 – Replace existing RHRSW pumps
- DCN 70946 – Modification to Emergency Equipment Cooling Water design requirements and calculations

b. Findings

No findings were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors witnessed and reviewed post-maintenance tests (PMT) 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 function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents. The inspectors witnessed and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors verified that problems associated with PMTs were identified and entered into the Corrective Action Program (CAP). Documents reviewed are listed in the attachment. This activity constituted eleven Post Maintenance Test inspection samples, as defined in Inspection Procedure 71111.19.

- 3D 4kV shutdown board loss of power to the Normal Voltage Available Relays (WO 117608777)
- 3-CKV-67-723 check valve replacement (WO 117638683)

- Core Spray System II Inboard and Outboard Injection Valve Logic Functional Test 3-SR-3.3.5.1.6(CS II I/O) (WO 116617043)
- 2-FCV-73-16 HPCI turbine steam supply valve stroke test (WO 117494970)
- Unit 3 Standby Liquid Control system (WO 116617149)
- Unit 3 MSIV stroke testing following refueling outage maintenance (WO 116617273)
- Unit 3 Source Range Monitors (SRMs) (WO 115949599)
- Unit 2 HPCI packing replacement on the 2-FCV-73-16 steam admission valve (WO 117530716)
- 2-FSV-064-34 suppression chamber to standby gas inboard isolation valve stroke testing (WOs 117644740 and 117632970)
- Unit 3 HPCI flowrate testing following refueling outage maintenance (WOs 116617202, 116798508)
- Unit 3 RHR Functional Testing of Loop II Inboard and Outboard Valve Logic and Interlocks (WOs 116617066 and 117653018)

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Unit 3 Refueling Outage 17

a. Inspection Scope

From February 20, through March 26, 2016, the inspectors examined the refueling outage activities to verify that they were conducted in accordance with Technical Specifications (TS), applicable plant procedures, and the licensee's outage risk assessment and management plans. The inspectors monitored critical plant parameters and observed operator control of plant conditions through Cold Shutdown (Mode 4), Refueling (Mode 5), Plant restart and power ascension through Startup (Mode 2) and Run (Mode 1). This activity constituted one Refueling and Other Outage Activities inspection sample. Some of the significant outage activities specifically reviewed and/or witnessed by the inspectors were as follows:

Outage Risk Assessment
Prior to the beginning of the refueling outage, the inspectors attended outage risk assessment team meetings and reviewed the Outage Risk Assessment Report. The inspectors reviewed the daily Refueling Outage Reports, including the Outage Risk Assessment Management (ORAM) Safety Function Status, and regularly attended the daily outage status meetings. The inspectors frequently discussed risk conditions and protected equipment with operations and outage management personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process
The inspectors witnessed the shutdown and cooldown of Unit 3 in accordance with applicable licensee procedures.

Decay Heat Removal
The inspectors reviewed licensee procedures for normal and alternate decay heat removal and conducted main control room panel and in-plant walkdowns of system and components to verify correct system alignment. During planned evolutions that resulted in increased outage risk conditions for shutdown cooling, inspectors verified that the plant conditions and systems identified in the risk mitigation strategy were available. In addition, the inspectors reviewed controls implemented to ensure that outage work was not impacting the ability of operators to operate spent fuel pool cooling, RHR shutdown cooling, and/or Alternate Decay Heat Removal system.

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

- Walked down selected safety-related equipment clearance and associated with tagout numbers:
    1) 3-TO-2016-005, Clearance 3-001-0005 for Main Steamline Plug Installation/Removal
    2) 3-TO-2016-005, Clearance 3-074-0007 for RHR System I Minimum Flow Valve while on Shutdown Cooling
    3) 3-TO-2016-003; Clearance 3-074-0049 for RHR System I repairs
    4) 3-TO-2016-003; Clearance 3-075-0028A for flow switch replacement on Core Spray System I
- Verified Reactor Coolant System (RCS) inventory controls, specifically, the makeup methods used during operations with the potential to drain the reactor vessel (OPDRV's)
- Verified electrical systems availability and alignment
- Observed the approach to and level controls during the reduced inventory condition needed for the reactor vessel flange repairs
- Observed the RCS hydro / leak test and simultaneous scram time testing.
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) and Technical Specification compliance during the various shutdown modes of operation, and mode transitions
- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations and overall integrity
- Examined foreign material exclusion controls particularly in proximity to and around the reactor cavity, equipment pit, and spent fuel pool
- Performed routine tours of the control room, reactor building, refueling floor and drywell
- Verified the licensee was managing fatigue by performing a sample review of fatigue assessments, schedules and work hours of online and outage personnel.

Reactor Vessel Disassembly and Refueling Activities
The inspectors witnessed selected activities associated with reactor vessel disassembly, and reactor cavity flood-up and drain down. The inspectors witnessed fuel handling operations during the reactor core fuel shuffles performed in accordance with Technical

Specifications and applicable operating procedures addressing refueling operations (in vessel), operations in the spent fuel pool, and fuel movement operations during refueling.

Drywell Closeout
The inspectors reviewed the licensee's conduct of Drywell Closeout, and performed a detailed closeout inspection.

Restart Activities
The inspectors specifically observed the following:

- Unit 3 approach to criticality and power ascension
- Reactor Coolant Heatup/Pressurization to Rated Temperature and Pressure

Corrective Action Program
The inspectors reviewed Condition Reports generated during the refueling outage and attended management review committee meetings to verify that initiation thresholds, priorities, mode holds, operability concerns and significance levels were adequately addressed. Resolution and implementation of corrective actions were also reviewed for completeness.

This activity constituted one Refueling and Other Outage Activities sample, as defined in Inspection Procedure 71111.20.

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. Documents reviewed are listed in the attachment. This activity constituted ten Surveillance Testing inspection samples: four routine tests, three in-service tests, and three containment isolation valve tests, as defined in Inspection Procedure 71111.22.

Routine Surveillance Tests:

- 0-GOI-300-1/ATT-15.22 Emergency Operating Instruction (EOI) Equipment Storage Box Inventory (Unit 2 Auxiliary Instrument Room)
- 3-SR-3.5.1.9(RHR I) Loop I RHR Simulated Automatic Actuation Test
- 3-SR-3.1.4.1 Unit 3 Scram Time Testing

- 3-SR-3.6.1.3.5(HPCI CM) HPCI Check Valve Operability Tests During Cold Shutdown

<u>In-service Tests</u>:
- Unit 3 Core Spray Loop I Discharge Relief Valve per 0-TI-577(TEST) Inservice Testing of ASME and Augmented Pressure Relief Devices
- 3-SR-3.5.1.6 (RHR I) Quarterly RHR System Rated Flow Test Loop I
- 2-SR-3.6.1.3.5(RCIC) RCIC System MOV Operability

<u>Containment Isolation Valve Tests</u>:
- 3-SR-3.6.1.3.5(64 RO) Primary Containment System Operability Test
- 3-SR-3.6.1.3.10(C) and (D) Main Steam Line C and D As Left Local Leak Rate Tests
- 3-SI-4.7.A.2.G-3/73B Primary Containment Local Leak Rate Testing HPCI Turbine Exhaust Penetration X-214

b. <u>Findings</u>

Unacceptable Preconditioning of RCIC Valve Prior to ASME In-Service Testing

<u>Introduction</u>: An NRC identified Green finding (FIN) was identified for the licensee's failure to meet TVA procedure NETP-116.3, "Inservice Testing Program Preconditioning Guidelines," because unacceptable preconditioning of the Unit 2 RCIC Steam Supply valve occurred prior to quarterly IST. Specifically, the preconditioning was unacceptable because the testing sequence was avoidable, it masked the actual as-found condition of the valve, and it could possibly result in an inability to verify the operability of the valve.

<u>Description</u>: On January 5, 2016, the licensee planned to perform RHR heat exchanger thermal performance testing on the 2B and 2D RHR heat exchangers by placing RHR in its suppression pool cooling mode and using RCIC exhaust steam into the suppression pool to help balance suppression pool water temperature. Prior to operating RCIC, the quarterly motor operated valve (MOV) operability IST per TS 5.5.6 was planned to be completed. A procedure error in the MOV test prevented stroke time testing the RCIC steam supply valve as-written. The steam supply valve opens to admit steam to the RCIC turbine on low reactor water level. Because the licensee had vendors standing by to support the RHR heat exchanger testing, the licensee desired to proceed with the RHR heat exchanger tests prior to resolving the procedure error. Operators recognized that running RCIC would precondition the steam supply valve by cycling the valve before the IST was completed, and requested engineering to evaluate whether the preconditioning was acceptable. Engineering concluded that the preconditioning was acceptable because the situation was bounded by a generic licensee evaluation that justified the infrequent practice of performing preventive maintenance prior to IST. Inspectors reviewed the generic evaluation and determined that the evaluation was intended to justify preventive maintenance that may randomly occur prior to quarterly IST and was not a suitable justification to deliberately allow the infrequent performance of tests out of sequence, when such operations could be avoided without negatively impacting personnel or plant safety. Additionally, inspectors identified that engineers did not complete the required preconditioning evaluation forms described in licensee procedure NETP-116.3 "Inservice Testing Program preconditioning Guidelines."

Completion of these forms would have caused engineers to challenge the basis for the request since the request was being made strictly for scheduling convenience, which was one of the considerations on the form. The inspectors concluded that the preconditioning was unacceptable because the testing sequence was avoidable, it masked the actual as-found condition of the valve, and it could possibly result in an inability to verify the operability of the valve.

Analysis: The inspectors determined that the stroking of the RCIC steam supply valve prior to as-found IST constituted unacceptable preconditioning and was a performance deficiency. TVA procedure NETP-116.3, Section 3.3.5, "Unacceptable Preconditioning," required that "unacceptable preconditioning shall not be performed." The performance deficiency was more-than-minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage) in that the preconditioning resulted in the loss of information used to ensure system capabilities between quarterly tests. This finding was evaluated in accordance with NRC IMC 0609, Appendix A, Exhibit 2 "Mitigating Systems Screening Questions," dated June 19, 2012. The inspectors determined the finding was Green because the finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its TS allowable outage time, did not result in a loss of function of non-TS equipment, and did not involve the loss of equipment or function specifically designed to mitigate an external event. The inspectors determined that the finding had a cross-cutting aspect in the Human Performance area of Consistent Process [H.13], because individuals did not complete the required preconditioning evaluation forms described in licensee procedure NETP-116.3.

Enforcement: This finding does not involve enforcement action because no violation of a regulatory requirement was identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as a FIN (FIN 05000260/2016001-01, Unacceptable Preconditioning of RCIC Valve Prior to ASME In-Service Testing)

EMERGENCY PREPAREDNESS

1EP6 Drill Evaluation (IP 71114.06)

.1 January 13, 2016, EP Radiological Emergency Plan (REP) training drill

a. Inspection Scope

The inspectors observed an EP REP training drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures on January 13, 2016. This drill was intended to identify any licensee weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the Simulated Control Room and the Technical Support Center, to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure, and licensee conformance with other applicable Emergency Plan Implementing Procedures.

The inspectors 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 in to the CAP, as appropriate.

b. Findings

No findings were identified

.2 February 3, 2016, Simulator Based EP Radiological Emergency Plan (REP) training drill

a. Inspection Scope

The inspectors observed a simulator based EP REP training drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures on February 3, 2016. This drill was intended to identify any licensee weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the Simulated Control Room, to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure, and licensee conformance with other applicable Emergency Plan Implementing Procedures. The inspectors 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 in to the CAP, as appropriate.

b. Findings

Introduction. The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50.54(q)(2), for the licensee's failure to maintain the effectiveness of its emergency plan by ensuring procedures for use by the emergency response organization are maintained and up-to-date as required by 10 CFR 50.47(b)(16). Specifically, the effectiveness of emergency plan implementing procedure EPIP–5, General Emergency, Revision 45, was reduced by the inadvertent omission of a portion of Section 3.6, which involved making Protective Action Recommendation (PAR) upgrades. The licensee's procedure change review process failed to identify these omissions. Additional minor inadvertent omissions were also identified by the inspectors.

Description. Following performance of Job Performance Measure (JPM) 679 conducted during an initial license examination, the inspectors questioned the facility on an applicant response to a PAR upgrade following a wind shift (i.e., applicant did not provide answers as specified by the key). The inspectors asked the licensee for their technical justification to support the answer originally proposed in the JPM. The licensee entered this request into the corrective action program (CAP) as condition report (CR) 1112692. Also, during the licensee's review of the JPM, it was identified that some procedural inadequacies existed related to how one progresses through the associated emergency plan implementing procedure (EPIP). This was entered into the CAP as CR 1106129.

Further inspection assistance was provided by regional emergency preparedness inspectors to determine if there was an issue regarding the methodology of determining PAR upgrades. It was then determined that in 2014, a wholesale change out of all EPIPs was performed. These changes included a re-formatting of the documents and migration to a template format for ease when performing future procedure changes. The inspectors then reviewed EPIP-5, General Emergency, Revisions 44 through 49, and identified several discrepancies, including the inadvertent removal of pertinent information that ultimately affected the licensee's ability to successfully demonstrate proficiency in PAR upgrades. The inspectors also identified that Revision 44 contained the appropriate language to determine PAR upgrades and affected EPZ sectors. EPIP-5, Revision 48 was in place and used during the administration of the license applicant's JPM. Neither Revision 48, nor its associated Appendix F, General Emergency Follow-Up Information Form, identified that all affected EPZ sectors be included in a PAR upgrade. The inspectors determined that Revision 45 was where the appropriate language started disappearing, and then Revision 48 was where all references to Appendix J, Upgrade – Protective Action Recommendation, were no longer contained in the procedure.

In October 2015, the inspectors identified a similar issue at another Tennessee Valley Authority (TVA) site and issued a Green NCV. However, the licensee's extent of condition (EOC) was limited in scope and only included a corporate document review and not a site specific review at each TVA nuclear site. This was a missed opportunity to identify discrepancies in the Browns Ferry EPIPs. Corrective actions to date included a revision (49) to EPIP-5, effective January 7, 2016, essentially replacing Section 3.6 and references to appropriate Appendices, and a broader scope EOC to review all site EPIPs to ensure no other inadvertent omissions were made.

Analysis. The licensee's failure to adequately maintain emergency plan implementing procedure EPIP-5, General Emergency, as required by 10 CFR 50.54(q)(2), was a performance deficiency. Specifically, the effectiveness of EPIP–5, Revision 45, was reduced by the inadvertent removal of portions of Section 3.6, which involved making Protective Action Recommendation upgrades. The inspectors determined that the performance deficiency was more than minor using NRC Inspection Manual Chapter (IMC) 0612, Appendix B, Issue Screening, because the performance deficiency was associated with the procedure quality attribute of the Emergency Preparedness (EP) cornerstone, adversely affected the associated cornerstone objective, and may have been used had an emergency been declared. The finding was evaluated using the EP significance determination process and was identified as having very low safety significance (Green) because it was a failure to comply with NRC requirements and was not a loss of the planning standard function. The finding was associated with a cross-cutting aspect in the Evaluation component of the Problem Identification and Resolution area because the licensee failed to thoroughly evaluate a similar issue at one of its other sites to ensure extent of conditions commensurate with their safety significance are thoroughly resolved. [P.2]

Enforcement. Title 10 CFR 50.54(q)(2) requires, in part, that a licensee authorized to possess and operate a nuclear power reactor shall follow and maintain the effectiveness of an emergency plan which meets the requirements in Appendix E to this part and the planning standards of 50.47(b). Title 10 CFR 50.47(b)(16) requires, in part, that responsibilities for plan development and review, and for distribution of emergency plans, which include emergency plan implementing procedures, are established. Contrary to the above, the licensee failed to maintain the effectiveness of its emergency plan by not ensuring a thorough review was conducted when revising EPIPs. Specifically, the effectiveness of emergency plan implementing procedure EPIP–5, General Emergency, Revision 45, was reduced by the inadvertent removal of a portion of Section 3.6, which involved making Protective Action Recommendation upgrades, and the procedure change review process failed to identify these omissions. The procedure change had been in place since September 2014, until January 2016, when a corrected revision was issued. The licensee entered the issue into their CAP as CR 1133821. Corrective actions implemented were to perform an extent of condition review of all site EP procedures and revise EPIP-5. Because this failure is of very low safety significance (Green) and has been entered into the licensee's CAP, this violation is being treated as a NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000259/2016001, 05000260/2016001, 05000296/2016001-02, "Failure to adequately maintain emergency plan implementing procedures."

These activities constituted completion of one Emergency Preparedness drill evaluation and one Simulator Based Emergency Preparedness drill evaluation, as defined in Inspection Procedure 71114.06. Documents reviewed are listed in the attachment.

2. RADIATION SAFETY (RS)

2RS1 Radiological Hazard Assessment and Exposure Control (71124.01)

a. Inspection Scope

Hazard Assessment and Instructions to Workers. During facility tours, the inspectors directly observed radiological postings and container labeling for areas established within the radiologically controlled area (RCA) of the Unit 1 (U1), Unit 2 (U2), and Unit 3 (U3) reactor buildings, U1,U2, and U3 turbine buildings, and radioactive waste (radwaste) processing and storage locations. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. The inspectors reviewed survey records for several plant areas including surveys for airborne radioactivity, gamma surveys with a range of dose rate gradients, surveys for alpha-emitters and other hard-to-detect radionuclides, and pre-job surveys for upcoming tasks. The inspectors also discussed changes to plant operations that could contribute to changing radiological conditions since the last inspection. The inspectors attended pre-job briefings and reviewed Radiation Work Permit (RWP) details to assess communication of radiological control requirements and current radiological conditions to workers.

Control of Radioactive Material. The inspectors observed surveys of material and personnel being released from the RCA using small article monitor, personnel contamination monitor, and portal monitor instruments. The inspectors discussed equipment sensitivity, alarm setpoints, and release program guidance with licensee staff. The inspectors also reviewed records of leak tests on selected sealed sources and discussed nationally tracked source transactions with licensee staff.

Hazard Control. The inspectors evaluated access controls and barrier effectiveness for selected High Radiation Area (HRA), Locked High Radiation Area (LHRA), and Very High Radiation Area (VHRA) locations and discussed changes to procedural guidance for LHRA and VHRA controls with Radiation Protection (RP) supervisors. The inspectors reviewed implementation of controls for the storage of irradiated material within the spent fuel pool. Established radiological controls, including airborne controls and electronic dosimeter (ED) alarm setpoints, were evaluated for selected Unit 3 Refueling Outage 17 tasks. In addition, the inspectors reviewed licensee controls for areas where dose rates could change significantly as a result of plant shutdown and refueling operations. The inspectors also reviewed the use of personnel dosimetry including extremity dosimetry and multibadging in high dose rate gradients.

Radiation Worker Performance and RP Technician Proficiency Occupational workers' adherence to selected RWPs and RP technician proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Jobs observed included maintenance and refueling activities in the drywell, reactor building, and refueling floor in high radiation and contaminated areas. The inspectors also evaluated worker responses to dose and dose rate alarms during selected work activities.

Problem Identification and Resolution The inspectors reviewed and assessed condition reports associated with radiological hazard assessment and control. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with licensee procedures. The inspectors also reviewed recent self-assessment results.

Radiation protection activities were evaluated against the requirements of Updated Final Safety Analysis Report (UFSAR) Section 12, Technical Specifications (TS) Sections 5.4 and 5.7, 10 CFR Parts 19 and 20, and approved licensee procedures. Licensee programs for monitoring materials and personnel released from the RCA were evaluated against 10 CFR Part 20 and IE Circular 81-07, "Control of Radioactively Contaminated Material". Documents and records reviewed are listed in the Attachment.

The inspectors completed the required seven samples specified in Inspection Procedure (IP) 71124.01.

b. Findings

Unauthorized entry into a high radiation area

Introduction: A self-revealing, Green, Non-cited Violation (NCV) of TS 5.7.1, was identified for a worker who entered an HRA without proper authorization. Specifically, the worker entered a HRA using an incorrect RWP and without being briefed on the radiological conditions.

Description: On August 18, 2015, an individual was performing roving fire watch duties in the reactor and turbine buildings. The worker was signed in on a general access RWP that did not allow entry to HRAs. When the worker entered the Radwaste Ventilation Equipment Room vestibule area to check a door, he encountered a recently posted HRA surrounding a box and several drums. Dose rates in the HRA ranged up to

150 mrem/hr at 30 cm from the box. In order to complete his assigned fire watch duties the worker entered the HRA and subsequently received an ED dose rate alarm. Upon completion of the fire watch route, the worker exited the RCA and reported to RP that an alarm had been received. The licensee took immediate corrective actions including RCA access restriction for the individual and initiation of an investigation of the event including surveys of the areas entered along the fire watch route.

Analysis: The inspectors determined that the worker's entry into a HRA without receiving authorization per TS 5.7.1 was a performance deficiency. This finding was determined to be greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Human Performance and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, workers who enter HRAs without knowledge of the radiological conditions in the area could receive unintended occupational exposures. The finding was not related to As Low As Reasonably Achievable (ALARA) planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. Therefore, the inspectors determined the finding to be of very low safety significance (Green). This finding involved the cross-cutting aspect of Human Performance, Procedural Adherence [H.8] because the event was a direct result of the worker's failure to adhere to requirements for HRA access.

Enforcement: Technical Specification 5.7.1 requires that access to HRAs be controlled by means of an RWP and entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. Contrary to this, on August 18, 2015, a licensee employee entered a posted high radiation area without proper RWP authorization and without being knowledgeable of the radiological conditions. Upon identification, the licensee immediately implemented RCA access restrictions for the individual and completed surveys of the areas entered by the individual. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's Corrective Action Program (CAP) as CR 1072343. (NCV 05000259/260/296/2016001-03, Unauthorized Entry into a High Radiation Area).

. Unposted High Radiation Area – Two examples

Introduction: A self-revealing, Green, NCV of 10 CFR 20.1902(b), with two examples, was identified for the failure to post HRAs. Specifically, HRAs within the Unit 2 (U2) Control Rod Drive (CRD) Rebuild Room and the U2 Reactor Water Cleanup (RWCU) Holding Pump Room were unposted for several months in 2015.

Description:

First Example: On April 21, 2015 an RP technician entered the CRD rebuild room to perform a survey for installation of scaffold and identified an unposted HRA located behind a shield wall due to a bag of trash with a dose rate of 300 mrem/hr at 30 cm. CR 1017294 was entered in the CAP to document the unposted HRA. Follow-up investigation of the unposted HRA determined that high rad trash bags and equipment were relocated from a posted HRA in the room and placed behind the shield wall in

January 2015. However, HRA postings and controls were not put in place after relocating the materials. The investigation also identified that on April 5, 2015 a machinist entered the CRD rebuild room and received an unanticipated ED dose rate alarm and that no follow-up survey was performed. CR 1023385 was entered in the CAP to document the ED alarm and lack of follow-up actions. The inspectors noted that the licensee had multiple opportunities to identify the unposted HRA, including an unanticipated dose rate alarm, over a period of several months.

Second Example: On December 29, 2015 an RP technician performing a routine survey in the U2 RWCU Holding Pump room identified an unposted HRA due to a hotspot on piping in the room with a dose rate of 150 mrem/hr at 30 cm. CR 1119944 was entered in the CAP to document the unposted HRA. Follow-up investigation for the CR identified that on August 11, 2015 and on September 14, 2015 surveys had been performed in the same area. Although these surveys also indicated that HRA conditions existed, a HRA posting was not listed on the survey form. The inspectors noted that, in each case, the RP technician who performed the survey and the approving supervisor failed to recognize the need for HRA postings.

The inspectors determined that these issues were self-revealing, although the licensee missed multiple opportunities to recognize them, therefore this finding is considered to be “self-revealing” rather than “licensee identified”.

<u>Analysis</u>: The inspectors determined that the failure to post HRAs as required by 10 CFR 20.1902(b), was a performance deficiency. This finding was determined to be greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Human Performance and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, failure to post and control high radiation areas can allow workers to enter HRAs without knowledge of the radiological conditions in the area and receive unintended occupational exposure. The finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The finding was not related to ALARA planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. Therefore, the inspectors determined the finding to be of very low safety significance (Green). This finding involved the cross-cutting aspect of Human Performance, Documentation [H.7] because the unposted high radiation areas were a direct result RP personnel failing to perform adequate review of survey data or recognize conditions that required additional radiological posting and control.

<u>Enforcement</u>: 10 CFR 20.1902(b) requires that the licensee post each high radiation area with a conspicuous sign or signs bearing the radiation symbol and the words ‘CAUTION, HIGH RADIATION AREA’ or ‘DANGER, HIGH RADIATION AREA”. Contrary to this, from January 1, 2015 to December 29, 2015, the licensee failed to post multiple HRAs with a conspicuous sign or signs bearing the radiation symbol and the words ‘CAUTION, HIGH RADIATION AREA’ or ‘DANGER, HIGH RADIATION AREA’. This violation is being treated as an NCV, consistent with Section 2.3.2 of the

Enforcement Policy. The violation was entered into the licensee's CAP as CR 1017294 and CR 1119944. (NCV 05000259/260/296/2016001-04, Unposted High Radiation Areas).

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

a. Inspection Scope

Waste Processing and Characterization. During inspector walk-downs, accessible sections of the liquid and solid radwaste processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included storage tanks, transfer piping, resin dewatering and packaging components, and abandoned radwaste processing equipment. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The inspectors reviewed the 2014 Annual Radioactive Effluent Report and radionuclide characterizations from 2015 - 2016 for selected waste streams. For Reactor Water Cleanup resin, filters, and Dry Active Waste (DAW), the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined quality assurance comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology were evaluated and discussed with radwaste staff. The inspectors also reviewed the licensee's process for monitoring changes in waste stream isotopic mixtures.

Radioactive Material Storage. During walk-downs of indoor and outdoor radioactive material storage areas, the inspectors observed the physical condition and labeling of storage containers and the posting of Radioactive Material Areas. The inspectors also reviewed licensee procedural guidance for storage and monitoring of radioactive material.

Transportation. The inspectors evaluated shipping records for consistency with licensee procedures and compliance with NRC and Department of Transportation (DOT) regulations. The inspectors reviewed emergency response information, DOT shipping package classification, waste classification, radiation survey results, and container handling methodology. The inspectors also observed shipment preparations for a DAW package and evaluated technician performance and knowledge of DOT requirements.

Problem Identification and Resolution. The inspectors reviewed condition reports in the areas of shipping and radwaste processing. The inspectors evaluated the licensee's ability to identify and resolve the issues.

Radwaste processing, radioactive material handling, and transportation activities were reviewed against the guidance and requirements contained in the licensee's Process Control Program, UFSAR Chapter 9, 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, the Branch Technical Position on Waste Classification (1983), and NUREG-1608

"Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated Objects". Documents reviewed during the inspection are listed in the report Attachment.

The inspectors completed the required seven samples specified in IP 71124.08.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR 71.5 for the failure to include the correct Proper Shipping Name (PSN) on radioactive material shipping papers in accordance with the requirements of DOT regulation 49 CFR 172.202. This resulted in multiple Low Specific Activity (LSA) shipments containing quantities exceeding an A_2 value being shipped as "UN2915, Radioactive Material, Type A Package".

Description: From January 14 to March 20, 2014, the licensee made several shipments of radioactive filters to a waste processing facility in Tennessee. Six of these shipments were made using Type A casks and included "UN2915, Radioactive Material, Type A Package" as the identification number and PSN in box 11 of NRC Form 540 (Shipping Paper). During a review of the records for these shipments, the inspectors noted that the packages actually contained quantities of radioactive material in excess of an A_2 value. This indicated that the shipments had exceeded DOT activity limits for Type A packagings and may have required more robust Type B casks. However, an exception to the Type A package activity limits is allowed for materials that meet the definition of LSA. One of the requirements to receive this exception is that the dose rate from the unshielded package contents (filter liner) is less than 1 R/hr at 3 meters. The inspectors determined that all six shipments met the requirements for the LSA exception, but noted that some of the shipping checklists had been marked "not applicable" when prompted to verify the unshielded 3-meter dose rate. Discussions with licensee shipping staff indicated that the requirements for LSA shipments, and when to use an LSA identification number and PSN (e.g. UN3321, Radioactive Material, LSA-II), were not well understood. The licensee documented this issue in CR 1145617. Licensee corrective actions included updating the software used to perform shipping activities and additional training of personnel.

Analysis: The inspectors determined that the failure to include the correct PSN on the shipping papers as required by DOT regulation 49 CFR 172.202 was a performance deficiency. The finding was greater than minor because it was associated with the Public Radiation Safety Cornerstone, Program & Process attribute (transportation program), and adversely affected the associated cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Using the Type A PSN for a package with radioactivity levels exceeding an A_2 value is an underrepresentation of the package contents and could lead to confusion for the receiving licensee, the driver, or an accident first responder. The significance of the finding was evaluated using the Public Radiation Safety Significance Determination Process. The issue involved transportation, but there were no radiation limits exceeded, and there was no package breach. In addition, it did not involve a Certificate of Compliance or low-level burial problem, nor was there a failure to make notifications or provide emergency response information. Therefore, the inspectors determined that the finding was of very low safety significance (Green). The finding has a cross-cutting aspect in the area of Human Performance, Training, because the DOT requirements pertaining to LSA shipments were not well understood. [H.9]

Enforcement: 10 CFR 71.5 requires the licensee to comply with the DOT regulations in 49 CFR Parts 170 through 189. The regulations in 49 CFR 172.202 require hazardous material shipping papers to contain the identification number and PSN as described in the Hazardous Material Table (49 CFR 172.101). A "Type A package" is defined in 49 CFR 173.403 as a Type A packaging with contents limited to an A_2 quantity of radioactive material. Contrary to the above, from January 14 through March 20, 2014, six shipments that exceeded an A_2 quantity of radioactive material were made using an identification number and PSN from the Hazardous Material Table that did not accurately describe the shipping package (UN2915, Radioactive material, Type A Package). Immediate licensee corrective actions included updating the software used to perform shipping activities and additional training of personnel. Because this violation was of very low safety significance and it was entered into the licensee's CAP (CR 1145617), this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000259, 260, 296/2016001-05; Failure to Include the Correct Proper Shipping Name on Radioactive Material Shipping Papers).

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Cornerstone: Barrier Integrity

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 first quarter 2015 through the fourth quarter of 2015. 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., PERs, 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 six PI inspection samples, as defined in Inspection Procedure 71151.

- Unit 1, 2, and 3 Reactor Coolant System activity
- Unit 1, 2, and 3 Reactor Coolant System leakage

b. Findings

No findings were identified.

.2 Radiation Protection

a. Inspection Scope

Occupational Radiation Safety Cornerstone The inspectors reviewed recent Occupational Exposure Control Effectiveness PI results for the Occupational Radiation Safety Cornerstone and reviewed PI records generated between April, 2015 – January, 2016. For the assessment period, the inspectors reviewed ED alarm logs and CRs related to controls for exposure significant areas. Documents reviewed are listed in the report Attachment. This activity constituted three PI inspection samples, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution of Problems (71152)

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

a. Inspection Scope

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 (MRC) and Plant Screening Committee (PSC) meetings.

b. Findings

No findings were identified.

.2 Focused Annual Sample Review – Control Air Compressor Trip on December 25, 2015:

a. Inspection Scope

The inspectors conducted a review of the circumstances surrounding the A and B control air compressors tripping on December 25, 2015 during a thunderstorm. The inspectors reviewed the FSAR to verify that this condition had been previously analyzed. The service air system responded as described in the FSAR to maintain control air pressure above 80 psig. The licensee has documented their review and corrective actions from this issue in CR 1119072. This activity constituted one focused annual inspection sample, as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

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

.1 (Closed) Licensee Event Report (LER) 05000259/2015-005-00 Inboard Main Steam Isolation Valve Actuators Inoperable for Longer Than Allowed by Technical Specifications

a. Inspection Scope

On October 29, 2015, the licensee determined that the Main Steam Isolation Valve (MSIV) accumulators on all BFN inboard MSIVs are of insufficient size to provide the MSIV actuators adequate air volume, at the required pressure, to close the MSIV during a Loss of Coolant Accident (LOCA). Therefore, availability of Drywell Control Air (DWCA) nitrogen from the Containment Inerting system or from the Containment Atmospheric Dilution system was determined to be necessary for operability of inboard MSIVs. From December 1, 2012, to the time of discovery, there were multiple occasions where BFN Unit 1, 2, or 3 DWCA systems were aligned to receive nitrogen from the Plant Control Air system, resulting in the inoperability of multiple MSIVs for longer than allowed by BFN Technical Specification Limiting Conditions for Operation 3.6.1.3, Condition A.

The inspectors reviewed the licensee event report dated December 28, 2015, the vendor report that provided the basis for the licensee's determination, and the licensee's operability analysis. The inspectors also reviewed the design basis documents for all the systems mentioned in the licensee event report. The inspectors also reviewed the licensee's corrective actions associated with this LER.

b. Findings

The enforcement aspects of this event are discussed in Section 4OA7.1. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 05000296/2014-003-00 Primary Containment Isolation Valve Inoperable for Longer Than Allowed by Technical Specifications

a. Inspection Scope

On June 2, 2014, the licensee determined that the Unit 3 RHR SDC Inboard Suction Valve Isolation relay failed to energize. Due to this relay failure, the RHR SDC Inboard Suction Valve would not automatically close in response to a Primary Containment Isolation System signal. For three time periods where this valve was in an open position without automatic closure capability, the plant was in a condition prohibited by Technical Specifications. The cause of the event was determined to be due to a human performance error that affected the wiring of the relay after it had been successfully post maintenance tested on March 7, 2014. The wiring of the relay was corrected on June 6, 2014.

b. Findings

1. Introduction: The NRC identified a Green NCV of TS 5.4.1, Procedures, for the licensee's failure to implement OPDP-8, Operability Determinations and LCO Tracking. Specifically, the licensee failed to track the applicability of condition 'A' of TS LCO 3.6.1.3 upon discovery of the equipment failure described in LER 05000296/2014-003-00.

   Description: As described in LER 05000296/2014-003-00, on June 2, 2014, the licensee made an operations log entry that TS LCO 3.3.6.1 conditions A, C, and F were entered due the failure of valve actuating relay 3-RLY-074-10A-K98A. The failure of this relay caused the outboard RHR SDC primary containment isolation valve (PCIV) to be inoperable since the valve would not automatically close upon receipt of a primary containment isolation signal. The appropriate TS LCO for an inoperable PCIV is TS LCO 3.6.1.3. However, operators instead entered TS LCO 3.3.1.6 because they believed the relay failure was strictly associated with the support system (primary containment isolation instrumentation). TS LCO 3.0.6 does allow entry into only the support system LCO in lieu of the supported system LCO when the degraded condition is solely associated with the support system; however, in this case, the relay failure did not actually adversely affect the operability of any instrumentation channels of the support system. The relay was associated with the valve actuation circuitry, which was not specifically covered under any action statements in TS LCO 3.3.1.6. Despite the error, the plant still met the TS LCO required actions for TS LCO 3.6.1.3, condition A, at the time of discovery, since the pathway was already isolated by a de-activated automatic valve per required action A.1 and because the valve was restored to an operable status prior to needing to re-verify isolation once every 31 days per required action A.2 of LCO 3.6.1.3.

   Analysis: The inspectors determined that the failure to track applicable technical specification action statements as required by section 3.5.1 of OPDP-8, "Operability Determination Process and Limiting Conditions for Operation Tracking" was a performance deficiency. This requirement was not satisfied because operators identified and tracked an incorrect TS LCO. The performance deficiency was more-than-minor because, if left uncorrected, would have the potential to lead to a more significant safety concern. Specifically, this failure was indicative of a programmatic weakness with the licensee's evaluation of certain logic circuit failures which resulted in misapplication of the allowances of TS LCO 3.0.6 and could lead to further inappropriate TS LCO entries and missed TS Actions. The inspectors determined that this type of error was likely to recur which could lead to more significant errors if uncorrected. The inspectors evaluated the significance of this finding using IMC 0609 Appendix A, dated June 19, 2012, The Significance Determination Process (SDP) for Findings at Power, Exhibit 3, Barrier Integrity Screening Questions. The inspectors determined that this finding was of very low safety significance (Green) because the error did not result in an actual open pathway in the physical integrity of reactor containment, containment isolation system or heat removal components. The performance deficiency had a cross-cutting aspect of Training in the area of Human Performance because the finding was indicative of a knowledge gap among the operations department (H.9).

Enforcement: TS 5.4.1.a, “Procedures,” required, in part, that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Section 1(h), “Administrative Procedures,” required procedures addressing log entries, which was partially implemented by OPDP-8, “Operability Determination Process and Limiting Conditions for Operation Tracking,” Revision 21. OPDP-8, section 3.5.1, required, in part, that operators make log entries of entry and exit from technical specification action statements. Contrary to the above, the licensee failed to make plant log entries for the entry and exit from TS LCO 3.6.1.3, primary containment isolation valves, condition “A” on June 2, 2014. Immediate corrective actions included entering this issue into their corrective action program as CR 1115172. Because this finding is of very low safety significance (Green) and was entered into the corrective action program, this violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. (NCV 0500296/2016001-06: Failure to Identify Applicable Technical Specification Action Statement for a PCIV)

An additional enforcement aspect from this event is discussed in Section 4OA7.2. This licensee event report is closed.

.3 (Closed) Licensee Event Report (LER) 05000296/2015-002-00 Switch Failure Rendered Automatic Startup of Some Emergency Core Cooling System Pumps Inoperable Longer than Allowed by Technical Specifications

a. Inspection Scope

On January 22, 2015, the licensee determined that the automatic start function for the Unit 3 Core Spray pumps 3B and 3D, Residual Heat Removal pump 3D, and the D1 Residual Heat Removal Service Water Pump were inoperable longer than allowed outage time. The cause was a failure to perform preventative maintenance as recommended by the manufacturer or pre-emptive replacement of the MJ(52STA) switches, allowing them to fail. The MJ(52STA) switches support the automatic start function of the pumps following an accident signal when normal power was maintained. The inspectors reviewed the LER dated April 20, 2015 and all associated CAP documents and causal analysis. The licensee used a procedurally allowed delay to determine that the previously mentioned pumps were inoperable longer than their allowed outage time.

b. Findings

The enforcement aspects of this event are discussed in Section 4OA7.3. This LER is closed.

.4 (Closed) Licensee Event Report (LER) 05000296/2015-005-00 Automatic Actuation of 3D Diesel Generator Due to 4kV Shutdown Board Trip During Testing

a. Inspection Scope

On August 20, 2015, while installing test equipment on the 3ED 4kV Shutdown Board (SD BD), for an online dynamic motor test of the 3D RHR pump motor, the Unit 3 Control Room received a degraded voltage alarms and under voltage alarms for the 3ED SD BD. The 3ED 4kV SD BD normal feeder breaker opened, and the 3D Emergency Diesel Generator (DG) fast started and tied onto the board. Troubleshooting discovered that two fuses had cleared in the 3ED SD BD. Because a definitive cause for the failures was not identified, the licensee developed corrective actions for the most probable causes associated with faulty test equipment, and human performance errors.

b. Findings

The Licensee Event Report was reviewed. No findings or violations of NRC requirements were identified. This LER is closed.

.5 (Closed) Licensee Event Report (LER) 05000259/260/296/2015-004-00 Containment Atmosphere Dilution B Train Supply System Inoperable Longer Than Allowed by Technical Specifications

a. Inspection Scope

On September 29, 2015, the TVA discovered a small puncture hole in a 2 inch stainless steel underground Containment Atmosphere Dilution (CAD) pipe. The cause and date of occurrence was unable to be determined by TVA. An engineering evaluation determined the B train of CAD would not have been able to provide its specified safety function. Based on the discovery the licensee concluded that Technical Specification LCO 3.6.3.1, Conditions A and C completion times would not have been met to place the units in Mode 3 within 12 hours. Additionally due to times when train A CAD was also inoperable the licensee concluded that there had been occasions when a loss of safety function in accordance with NUREG-1022 also occurred. As corrective action TVA replaced the damaged pipe and created an action to perform a piping integrity test of the CAD system. The inspectors reviewed the LER dated April 20, 2015 and all associated CAP documents and causal analysis.

b. Findings

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

This activity constituted completion of five event follow-up samples, as defined in Inspection Procedure 71153. Documents reviewed are listed in the attachment.

4OA6 Meetings, Including Exit

On April 19, 2016, 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 verified that all proprietary information was returned to the licensee.

4OA7 Licensee-Identified Violations

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

1. Licensee Event Report (LER) 05000259/2015-005-00 Inboard Main Steam Isolation Valve Actuators Inoperable for Longer Than Allowed by Technical Specifications. TS 3.6.1.3 condition A required, in part, that when one or more penetration flow paths with one Primary Containment Isolation Valve (PCIV) inoperable except due to MSIV leakage not within limits that within 4 hours the affected penetration flow path be isolated by use of at least one closed and de-activated automatic valve with flow through the valve secured. TS 3.6.1.3 condition E required, in part, that when the Required Action and associated Completion Time of Condition A was not met in MODE 1, that the Unit must be placed in Mode 3 within 12 hours and Mode 4 within 36 hours. Contrary to the above, on multiple occasions between December 1, 2012 and October 29, 2015, the inboard MSIV's PCIV function was inoperable on all main steam lines on all three Units longer than the allowed outage time and the follow on action completion time. This violation is documented in the licensee's CAP as CR 1098857. This finding was screened to Green using IMC 0609 Appendix H dated May 6, 2004. Table 6.2 Phase 2 Risk Significance was used to screen the finding to Green because at no point during the time period between December 1, 2012 and October 29, 2015 did any outboard MSIV leakage on any Unit exceed 10,000 scfh.

2. Licensee Event Report (LER) 05000296/2014-003-00 Primary Containment Isolation Valve Inoperable for Longer Than Allowed by Technical Specifications 10 CFR 50, Appendix B, Criterion 5 required, in part, that activities affecting quality be implemented in accordance with documented procedures and drawings. Contrary to the above, between March 7, 2014 and June 6, 2014, relay 3-RLY-074-10A-K98A was wired incorrectly as discussed in LER 05000296/2014-003-00. The licensee corrected the wiring and entered the issue into the licensee's corrective action program as CR 892500. Inspectors screened the violation using IMC 0609, Appendix G, Attachment 1, Exhibit 3 "Mitigating Systems Screening Questions," dated May 9, 2014. Because the finding degraded a functional auto-isolation of RHR on low reactor water level, a Phase 2 screening was required. Using attachment 3, "Phase 2 Significance Determination Process Template for BWR During Shutdown," dated February 28, 2005, inspectors completed Worksheet 1 for "Loss of Inventory in Plant Operating State 1 (Head On)" and determined the risk was approximately 1e-7/yr, which was less than the 1e-6/yr threshold for a greater than Green finding. The dominant core damage sequence was the failure to isolate a reactor coolant leak and subsequent failure by operators to open vent paths (e.g. a safety relief valve) to control RCS pressure to enable continued low pressure injection. In the evaluation, no operator recovery credit was given for leak isolation, but credit was given for the redundant isolation valve that was operable which could have satisfied the automatic isolation function. The Regional Senior Reactor Analyst performed a detailed risk review of the finding. The risk review considered both the outage related risk, and the risk associated with a trip from power that would have the plant in shutdown cooling during the recovery. A screening analysis using bounding assumptions and the risk models ISL-RHR event tree was performed. The dominant cutsets involved failure of the redundant valve to operate, and operator actions to recover. Because of the short exposure time

during the shutdown periods, the redundant valve with the automatic action available, and the availability of operator recovery, the Finding was determined to be Green. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy.

3. Licensee Event Report (LER) 05000296/2015-002-00 Switch Failure Rendered Automatic Startup of Some Emergency Core Cooling System Pumps Inoperable Longer than Allowed by Technical Specifications: TS 3.3.5.1 condition A required, in part, that when one or more channels of Emergency Core Cooling System (ECCS) Instrumentation were inoperable that the condition listed in table 3.3.5.1-1 be immediately entered for that channel. MJ(STA 52) switch on breaker BFN-3-BKR-211-03ED/008 failed rendering automatic start sequence timing for the 3B and 3D Core Spray pumps, the 3D RHR Pump, and the D1 RHRSW Pump sequence time to become inoperable for conditions where normal power was maintained. This resulted in the licensee not meeting the TS completion times from September 17, 2014 until January 24, 2015, for TS 3.3.5.1 condition C (Core Spray Pumps 3B and 3D), TS 3.5.1 condition B (3D RHR pump), and TS 3.7.1 condition G (D1 RHRSW pump). This licensee identified violation is documented in the licensee's CAP as CR 980277. This finding was able to be screened to Green using IMC 0609 Appendix A dated June 9, 2012 because although these pumps were inoperable, their respective systems did not lose their function as emergency starts were not affected.

4. Licensee Event Report (LER) 05000259/260/296/2015-004-00 Containment Atmosphere Dilution B Train Supply System Inoperable Longer Than Allowed by Technical Specifications: Technical Specification LCO 3.6.3.1, Containment Atmosphere Dilution System, Condition B required that when Two CAD subsystems are inoperable that the licensee verify by administrative means that the hydrogen control function is maintained and to restore one CAD subsystem to OPERABLE status within 7 days. Condition C required action to place the affected unit in Mode 3 within 12 hours if the Condition B completion time was not met. Contrary to Technical Specification LCO 3.6.3.1 condition C, completion times were not met to place the units in Mode 3 within 12 hours when both trains of CAD were considered unavailable. This licensee identified violation is documented in the licensee's CAP as CR 1087766. This finding was screened to Green using IMC 0609 Appendix H,Table 6.1 because the finding did not affect any of the listed Systems, Structures, or Components important to LERF.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
K. Bronson, Senior Site Vice President
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
B. Tidwell, EP Manager
H. Smith, Fire Marshal
M. Lawson, Radiation Protection Manager
Q. Leonard, System Engineering Manager
D. Campbell, Superintendent of Operations
M. Kirschenheiter, Assistant Director for Site Engineering
J. Polickoski, Senior Corporate Licensing Project Manager
L. Slizewski, Operations Shift Manager
C. Whitworth, Operations Shift Manager
R. Loggins, Operations Shift Manager
M. Oliver, Licensing Engineer
E. Bates, Licensing Engineer
M. Acker, Licensing Engineer
R. Guthrie, System Engineer
J. Smith, System Engineer
J. Lacasse, System Engineer
D. Jackson, System Engineer
P. Campbell, System Engineer
L. Holland, System Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Opened and Closed

05000260/2015004-05	NCV	Failure to Properly Install the Unit 2 HPCI Turbine Steam Admission Valve Packing (Section 1R15)
05000260/2015004-06	NCV	Failure to Identify Significant Steam Leak on the Unit 2 HPCI Turbine Steam Admission Valve (Section 1R15)
05000260/2016001-01	FIN	Unacceptable Preconditioning of RCIC Valve Prior to ASME In-Service Testing (1R22).
05000259, 260, 296/2016001-02	NCV	Failure to adequately maintain emergency plan implementing procedures (1EP6.2)
05000259, 260, 296/2016001-03	NCV	Unauthorized Entry into a High Radiation Area (Section 2RS1)
05000259, 260, 296/2016001-04	NCV	Unposted High Radiation Areas (Section 2RS1)
05000259, 260, 296/2016001-05;	NCV	Failure to Include the Correct Proper Shipping Name on Radioactive Material Shipping Papers (2RS8)
05000296/2016001-06	NCV	Failure to Identify Applicable Technical Specification Action Statement for a PCIV (Section 4OA3.2)

Closed

05000260/2015004-05	AV	Failure to Properly Install the Unit 2 HPCI Turbine Steam Admission Valve Packing (Section 1R15)
05000260/2015004-06	AV	Failure to Identify Significant Steam Leak on the Unit 2 HPCI Turbine Steam Admission Valve (Section 1R15)
05000259/2015-005-00	LER	Inboard Main Steam Isolation Valve Actuators Inoperable for Longer Than Allowed by Technical Specifications (Section 4OA3.1)

05000296/2014-003-00	LER	Primary Containment Isolation Valve Inoperable for Longer Than Allowed by Technical Specifications (Section 4OA3.2)
05000296/2015-002-00	LER	Switch Failure Rendered Automatic Startup of Some Emergency Core Cooling System Pumps Inoperable for Longer Than Allowed by Technical Specifications (Section 4OA3.3)
05000296/2015-005-00	LER	Automatic Actuation of 3D Diesel Generator Due to 4kV Shutdown Board Trip During Testing (Section 4OA3.4)
05000259/260/296/2015-004-00	LER	Containment Atmosphere Dilution B Train Supply System Inoperable Longer Than Allowed by Technical Specifications (Section 4OA3.5)

Discussed
None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

0-AOI-100-7, Severe Weather, Rev 36
NPG-SPP-09.17, Temporary Equipment Control, Rev 6
0-GOI-200-1, Freeze Protection Inspection, Rev 81

Other Documents

Operator logs and equipment out of service documentation from February 2, 2016
Freeze Protection Open Work Order listing dated February 12, 2016
CR 1142631 Discontinue the use of the Cal Rod 480 volt space heaters that do not meet NPG-SPP-09.17 requirements
CR 1125334 Investigation into how 480 volt Cal Rod Heater got face down on the floor

Section 1R04: Equipment Alignment

Procedures

3-OI-64, Primary Containment System, Rev 59
0-GOI-300-3/ATT-1, Locked Valve Audit, Rev 169
2-SR-3.5.3.2, Reactor Core Isolation Cooling System Monthly Valve Position Verification, Rev 12
2-OI-71/ATT-1, Reactor Core Isolation Cooling Valve Lineup Checklist, Rev 61
2-OI-71/ATT-2, Reactor Core Isolation Cooling Panel Lineup Checklist, Rev 62
2-OI-71/ATT-3, Reactor Core Isolation Cooling Electrical Lineup Checklist, Rev 60
2-OI-71/ATT-4, Reactor Core Isolation Cooling Inspection Checklist, Rev 59
0-GOI-300-1/ATT-5, Unit 1 Reactor Building Operator Round Logs, Rev 254
0-GOI-300-1/ATT-7, Unit 2 Reactor Building Operator Round Logs, Rev 226
0-GOI-300-1/ATT-9, Unit 3 Reactor Building Operator Round Logs, Rev 251
1-OI-78, Spent Fuel Pool Cooling and Cleanup System Operating Instruction, Rev 68
3-OI-78, Spent Fuel Pool Cooling and Cleanup System Operating Instruction, Rev 61

Drawings

2-47E813-1, Flow Diagram Reactor Core Isolation Cooling System, Rev 59
3-47E855-1, Flow Diagram Spent Fuel Pool Cooling System, Rev 26

Other Documents

Operator logs and equipment out of service documentation from January 25, 2016
TS 3.6.3.2 and Basis for TS 3.6.3.2, Amendment 212
FSAR Section 4.7, Reactor Core Isolation Cooling System
DCN 67033, Rev A, Impulse Steam Trap Modification
BFN-50-7071, Reactor Core Isolation Cooling System General Design Criteria Document, Rev 22
OPL171.040, Reactor Core Isolation Cooling System, Operator Training, Rev 24
CR 1122875, Broken locking tab on RCIC steam trap outlet shutoff valve
CR 1122889, Missing locking chain on RCIC steam trap outlet shutoff valve
System Health Report for Reactor Core Isolation Cooling System 10/1/15 – 1/31/2016
CR 1146299, Unit 3 Control Rod Blade
Prompt Investigation Results for CR 1146299
WOs 112191561, 09-727605-000 for the last (2012) Unit 3 Torus Desludging and Coating repairs. (Divers)
CR 1150101 for 0-TI-417 Torus Coating inspection results

Section 1R05: Fire Protection

Procedures

NPG-SPP 18.4.7 Control of Transient Combustibles, Rev 5
0-GOI-200-1, Freeze Protection Inspection, Rev 81
NPG-SPP-18.4.8, Control of Ignition Sources (Hot Work), Rev 5
NPG-SPP-09.17, Temporary Equipment Control, Rev 6

Other Documents

Fire Protection Report Volume 1, Rev 20
Fire Protection Report Volume 2, Rev 52
CR 1143888 Evaluate Fire Watch Requirements for Salamander Heaters Used for Traveling Water Screen Freeze Protection

Section 1R06: Flood Protection Measures

Procedures

Browns Ferry PM 67718 Evaluation for Sump Pump Check for Handholes 15 and 26

Other Documents

NDN-000-999-2007-0031, IF – BFN Probabilistic Risk Assessment – Internal Flooding Analysis, Rev 0
CR 1144474 NRC Questions regarding TVA Calculation NDN-000-999-2007-0031

Section 1R07: Heat Sink Performance

Procedures

0-TI-322 Heat Exchanger Performance Testing, Rev 000

Other Documents

Appendix 39 to TVA Extended Power Uprate License Amendment Request, Heat Exchanger K Values Utilized in EPU Containment Analysis
WO 117064509 Unit 1 1B and 1D RHR HX thermal performance testing
Preliminary results for the Unit 1 1B and 1D RHR HX thermal performance testing

Section 1R08: Inservice Inspection Activities

Procedures:

N-VT-3, Visual Examination of Weld Ends, Fit-Ups, and Dimensional Examination of Weld Joints, Revision 30
N-VT-3, Visual Examination of Weld Ends, Fit-Ups, and Dimensional Examination of Weld Joints, Revision 27
N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Revision 38
N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Revision 37
N-MT-6, Magnetic Particle Examination for ASME and ANSI Code Components and Welds, Revision 35
N-MT-6, Magnetic Particle Examination for ASME and ANSI Code Components and Welds, Revision 34

N-UT-64, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Revision 16

Drawings:
3-47B455-631,Weld Map High Pressure Coolant Injection System
3-CHM-2413-C, High Pressure Coolant Injection System Support Locations, Revision 2

Work Orders/Work Requests:
WO 115569837, NOI U3R16-003 Remove MT Indication Identified on 3-47B455-631-IA
WO 116930078, Replace BFN-3-SHV-067-0557 ASME R&R Activity

Condition Reports:
CR845072, 3-CKV-67-736 failed Radiography
CR887989, Unit 1, Code Pressure Tests Deferred beyond Required Dates
CR908684, DG 3B indication of undercut in weld on 3B1 heat exchanger
CR908203, Pinhole leak on EECW oulet from the C DG Engine Cooler HX. ASME Code Class 3 piping
CR948169, Snubbers installed in incorrect location
CR1009747, BFN-2-SHV-074-0852 leaking thru (repair) and needs cap installed, 120 dpm

Miscellaneous Documents:
Welder Performance Qualification Test Record for Welders: J. Gautney, M. Tompkins, M. Briggs, B. Lawrence, and C. Newberry
Welding Procedure Qualification Record Nos: GTA18-B-1, GT11-0-1A, and GT11-SPEC-1
Detailed Welding Procedure Specification Nos: SM11-B-1-N, GTSM11-O-1-N, and GT18-O-1-N
Certified Material Test Report PO#: 516373-1, 415142, D65-64348, 904138, D65-94034, and 904807
Magnetic Particle Examination Report Nos : MT-14-006, MT-14-017, MT-14-018, MT-14-020, and MT-14-021
Visual Examination Report No: VT-14-062

Section 1R11: Licensed Operator Requalification
Other Documents
Unit 2 Browns Ferry Training Plan OPL 175S.039 Revision 0

Section 1R12: Maintenance Effectiveness
Procedures
0-TI-346 Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting – 10CFR50.65, Rev 47

Other Documents
RHR System Health dated March 15, 2016
10 CFR 50.65(a)(1) evaluations for RHR performed in January 2015 and September 2015
Preventative Maintenance plans for RHR for the next 12 months beginning March 1, 2016
CR 1143944, RHRSW pump vibration trending into alert range for D2 and A3 pumps

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
NPG-SPP-09.11.1 Equipment Out of Service Management, Rev. 10
NPG-SPP-07.3.4 Protected Equipment, Rev. 2
WO 117608777 3D 4kV Shutdown Board Normal Voltage Available relays de-energized

Drawings
3-45E766-15 Unit 3 Wiring Diagram 4160V shutdown auxiliary power, Rev 11

Other Documents
Browns Ferry Unit 1, 2, and 3 Equipment Out Of Service Report dated January 5-8, 2016, January 26, 2016, January 28, 2016 and February 23, 2016
eSOMS Action Tracking Status for Units 1, 2 and 3 on January 5-8, 2016, January 26, 2016, January 28, 2016 and February 23, 2016
eSOMS Narrative Logs dated January 5-8, 2016, January 26, 2016, January 28, 2016 and February 23, 2016
CR 1129086 Loss of Bypass Valves on Unit 3
CR 1140776 3D 4kV Shutdown Board Normal Voltage Available relays de-energized
TS 3.7.5 and basis for TS 3.7.5, Amendment 245
FSAR Section 11.5 Turbine Bypass System and Chapter 14 Plant Safety Analysis, Amendment 26
Thermal Limit Report for Unit 3 dated January 26, 2016

Section 1R15: Operability Evaluations

Procedures
1-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Systems, Rev 5
0-SR.3.8.1.A.1, Verification of Offsite Power Availability to 4.16 kV Shutdown Boards, Rev 14
2-SR-3.3.6.1.3(3D), HPCI Steam Line Space High Temperature Calibration (WO 115753595)
2-ARP-9-3F, Unit 2 Panel 9-3 Alarm Response Procedure, Rev 33
2-OI-73, HPCI System Operating Instruction, Rev 96
2-EOI-1, Unit 2 Flowchart Reactor Pressure Vessel Control, Rev 16
2-EOI-3, Unit 2 Flowchart Secondary Containment Control, Rev 16
3-SR-3.5.1.1(HPCI), Maintenance of Filled HPCI Discharge Piping, Revs 8 and 9
3-SI-3.2.4(DG D), EECW Check Valve Test on Diesel Generator D

Drawings
2-47E491-812 Mechanical Service Water, Air and Fire Protection for Elevation 519 ft. Unit 2 Reactor Building, Rev 2
0-47E455-6, Units 2 and 3 Mechanical High Pressure Coolant Injection System, Rev 3
3-47W455-8, Unit 3 Mechanical High Pressure Coolant Injection System, Rev 11

Other Documents
TS 3.5.1 and TS 3.5.1 Basis
TS 3.6.1.3 and TS 3.6.1.3 Basis
TS 3.6.2.3 and TS 3.6.2.3 Basis
TS 3.6.2.4 and TS 3.6.2.4 Basis
TS 3.8.1 and TS 3.8.1 Basis

TS 3.8.7 and TS 3.8.7 Basis
CR 1125899
CR 1126697
CR 1137399
FSAR Section 5.3, 7.3, 8.4, and 8.5, Amendment 26
Operator Logs dated January 16, 2016 and February 16, 2016
PDO for Unit 2 HPCI Steam Admission Valve (CRs 1127169, 1127172, and 1127173), Revs 0, 1, and 2
TVAEBFN052-CALC-001 BFN Unit 2 HPCI Room GOTHIC Analysis, Revs 0, 1, and 2
NDQ2999970011 Reactor Building Environmental Analysis for HELBs – Power Uprate, Rev 7
Root Cause Report for CR 1114188 HPCI 2-FCV-73-0016 Packing Failure, Rev 0
EOIPM Section 0-II-S Secondary Containment Control Bases, Rev 1
PDO for CR 1126697 3A RHR pump handswitch failure
Equipment Apparent Cause Evaluation for CR 1126697
CR 1093075
CR 1121668
PDO for CR 1038235, HPCI Venting
EWR09MEB073033, Flow rate and time required to vent gases from the highpoint HPCI discharge piping, dated November 19, 2009
WO 116798546
MDQ0000672013000125, Evaluation of EECW Component Flow Rates with the Most Limiting Pump Configuration, Rev 1
PDO for CR 1145025
WO 117638683 Conditional STS for 3-SI-3.2.4 (DG D)
CR 1127554
POE for CR 1127554

Section 1R18: Plant Modifications
Drawings
DWG 0-37W205-5, Pump Anchorage Drawing, Rev 8

Other Documents
DCN 71313, Replace Existing RHRSW Pumps with New Pumps that Have Closed Impellers, Upgraded Materials and a Different Bowl Assembly
SQUG Generic Implementation Procedure (GIP-3A)
CDQ0009992014000270, Seismic Qualification Analysis for the Sulzer Supplied RHRSW Pumps, Rev 0
CDQ0999-960096, Appendix E, EQE Calculation 50147-C-005, Resolution of USI A-46/IPEEE Seismic Programs at BFN Units 2 & 3, Rev 7
CDQ-0303-885699, Water Supply Pumping Station Concrete Structure Design, Rev 13
EQE Report 50147-R-001, Browns Ferry Nuclear Plant USI A-46 Seismic Evaluation Report, Rev 0
BFN-50-C-7106, Browns Ferry Nuclear Plant – Equipment Seismic / Structural Qualification (ESQ), Rev 5
MDQ0067910008, Flow Requirements for EECW-fed Components, Rev 18
MDQ0082000016, Diesel Generator Jacket Water Cooler Capacity and Tube Plugging, Rev 2

MDQ0000672013000125, Evaluation of EECW Component Flow Rates with the Most Limiting Pump Configuration, Rev 1
BFN-50-7082, Standby Diesel Generator General Design Criteria Document, Rev 24

Section 1R19: Post Maintenance Testing

Procedures
WO 117608777 3D 4kV Shutdown Board Normal Voltage Available relays de-energized
3-SR-3.8.1.9(3C) EDG 3C Emergency Load Acceptance Test, Rev 22
WO 117638683 3-CHV-67-723 replacement
3-SI-3.2.4(DG 3D) EECW Check Valve Test on EDG 3D, Rev 12
WO 116617043 Core Spray System II Inboard and Outboard Injection Valve Logic Functional Test 3-SR-3.3.5.1.6(CS II I/O)
WO 117494970 HPCI Sys Motor Operated Valve Operability
2-SR-3.6.1.3.5(HPCI) HPCI System Motor Operated Valve Operability, Rev 35
WO 116617149 Unit 3 Standby Liquid Control system functional test, Rev 33
3-SR-3.5.1.7 HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure
3-SR-3.5.1.8 HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at 150psig Reactor Pressure
SR 3.6.1.3.6 Main Steam Isolation Valve Fast Closure Test
SR-3.6.1.3.10(C) and (D) Primary Containment Local Leak Rate Test Main Steam Lines C and D
3-SR-3.3.5.1.6(C II I/O) Functional Testing of RHR Loop II Inboard and Outboard Valve Logic and Interlocks, Rev 7

Drawings
3-45E766-15 Unit 3 Wiring Diagram 4160V shutdown auxiliary power, Rev 11
3-45E768-8 Unit 3 Emergency Equipment Diesel Generator 3D Schematic Diagram, Rev 17

Other Documents
CR 1140776 3D 4kV Shutdown Board Normal Voltage Available relays de-energized
CR 1145025 EECW Flow below acceptance criteria on 3D EDG heat exchanger
CR 1146714 Procedure Change Request for 3-SR-3.3.5.1.6(CS II I/O)
TS 3.3.1.2 and TS 3.3.1.2 Basis, Source Range Monitors, Amendment 213
WO 115949599 SRM Dry Tube Replacement
WO 116638170 3-SR-3.3.1.2.4 SRM Signal to Noise Ratio Check
WO 116638194 3-SR-3.3.1.2.5 and 3-SR-3.3.1.2.6 SRM Functional Test with Reactor Mode Switch not in Run Position
WO 117644740 2-SR-3.6.1.3.5(64)
WO 117632970 Support engineering valve stroke test
WO 116617066 3-SR-3.3.5.1.6(C II I/O) – FT RHR Loop II Inboard and Outboard Valve Logic and Interlocks
WO 117653018PMTI for DCN 71214-10
CR 1146572
CR 1146603

Section 1R20: Refueling and Other Outage Activities
Drawings
3-45-E7 63-5, Recirc Pump start circuitry

Other Documents
Control Bay Habitability Zone (CBHZ) Breach Permit for WO 116591563
Tagout 3-TO-2016-005, Clearance 3-001-0005 for Main Steamline Plug
Tagout 3-TO-2016-005, Clearance 3-074-0007 for RHR System I Minimum Flow Valve
Tagout 3-TO-2016-003, Clearance 3-074-0049 for RHR System I repairs
Tagout 3-TO-2016-003, Clearance 3-075-0028A for flow switch replacement on Core Spray System I
WO 117684570, Lifted lead to allow Recirculation Pump bypass of Discharge Valve interlock
Hours worked records for 25 covered workers

Section 1R22: Routine Surveillance
Procedures
0-GOI-300-1/ATT-15.22 EOI Equipment Storage Box Inventory No. 2-EOI-000-0001 (U2 Auxiliary Instrument Room), Rev 209
2-EOI-Appendix-16L Bypassing HPCI High Temperature Isolation, Rev 2
2-EOI Appendix-5D Injection System Lineup HPCI, Rev 9
3-SR-3.6.1.3.5(64 RO), Primary Containment System PCIV Operability Test (WO 116618166)
0-TI-577(TEST) Inservice Testing of ASME and Augmented Pressure Relief Devices, Rev 7
0-TI-577 Inservice Testing of Pressure Relief Devices, Rev 6
3-SR-3.5.1.9(RHR I) – Loop I RHR Simulated Automatic Actuation Test (WO 116617167)
3-SR-3.5.1.6 (RHR I) Loop 1 RHR Quarterly Flowrate Inservice Test (WO 116798477)
3-SR-3.1.4.1 Unit 3 Scram Time Testing (WO 116798419)
SR-3.6.1.3.10(C) and (D) Primary Containment Local Leak Rate Test Main Steam Lines C and D
3-SI-4.7.A.2.G-3/73B Primary Containment Local Leak Rate Test HPCI Turbine Exhaust Penetration X-214, Rev 10
2-SR-3.6.1.3.5(RCIC) RCIC System MOV Operability, Revs 37 and 38
2-SR-3.5.3.3 RCIC System Rated Flow at Normal Operating Pressure, Rev 64
NPG-SPP-09.1.23 Inservice Testing Program Preconditioning Guidelines, Rev 0
NETP-116.3 Inservice Testing Program Preconditioning Guidelines, Rev 1
3-SR-3.6.1.3.5(HPCI CM) HPCI Check Valve Operability Tests During Cold Shutdown, Rev 3

Drawings
3-47E814-1, Flow Diagram Core Spray System, Rev 34

Other Documents
CR 1142864 3-FCV-64-17 closing time
CR 1143086 3-FCV-64-19 closing time
TS 3.6.1.3, Amendment 212
CR 1145197 3-RFV-75-543B failed to meet set pressure criteria
ASME OM Code-2004
FSAR Section 4.8 RHR System, Amendment 26
FSAR Section 6.4.4 LCPI System, Amendment 26

WO 116153448 3-SR-3.6.1.3.10(D) – [As Left] Main Steam Line D: Penetration X-7D
WO 117313230 3-SR-3.6.1.3.10(C) – [As Left] Main Steam Line D: Penetration X-7C
WO 116114560 3-SI-4.7.A.2.G-3/73B [As Found] HPCI Turbine Exhaust Penetration X-214
WO 117647047 3-SI-4.7.A.2.G-3/73B [As Found] HPCI Turbine Exhaust Penetration X-214
WO 116153298 3-SI-4.7.A.2.G-3/73B [As Left] HPCI Turbine Exhaust Penetration X-214
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Section 1EP6: Drill Evaluation (IP 71114.06)

Procedures
TVA Radiological Emergency Plan, Rev 106
Browns Ferry EPIP-1 Emergency Classification Procedure, Rev 52

Other Documents
Drill Guides for the January 13, 2016 and February 3, 2016 EP drills
Drill Evaluations for the January 13, 2016 and February 3, 2016 EP drills

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures, Guidance Documents, and Manuals
NPG-SPP-05.1, Radiological Controls, Rev. 0005
NPG-SPP-05.16, Radiological Controls for Performance of Radiography Operations, Rev. 0006
NPG-SPP-05.18, Radiation Work Permits, Rev. 0000
RCDP-8, Radiological Instrumentation/Equipment Controls, Rev. 0007
RCDP-17, Radiological Postings, Rev. 0000
RCI-1.2, Radiation, Contamination, and Airborne Surveys, Rev. 0029
RCI-7, Byproduct and Source Material Control, Rev. 0024
RCI-1.1, Radiation Operations Program Implementation, Rev. 0164
RCI-17, Control of High Radiation Areas and Very High Radiation Areas, Rev. 0084
RCI-23, Hot Spot Tracking Program, Rev. 0012
RCI-33, Diving Operations on the Refuel Floor, Rev. 0011
RCI-47, Diving Operations in the Radiologically Controlled Area, Rev. 0003

Records and Data
Air Sample Detail Report, for the period 1/1/2016 to 2/28/2016
BFN Key Control Record, RCI-17 Attachment 3, 1/1 - 3/3/2016
BFN Locked High Radiation Area Verification, RCI-17 Attachment 4, 3/1/2016
BFN Sealed Source Inventory List (Excel Worksheet), 2/25/2016
Browns Ferry Airborne Survey Logs for the period 1/1 to 2/28/2016
Browns Ferry PCE Log, PCE # 2016001 thru 2016008, 2/4 to 3/1/2016
Browns Ferry Unit 1, National Source Tracking System (NSTS) 2015 Confirmation of Annual Inventory Reconciliation, 1/22/2016
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Control of SFSPs [Unit 2 Spent Fuel Pool Inventory], 0-TI-540 Attachment 1, 7/31/2015
Control of SFSPs [Unit 3 Spent Fuel Pool Inventory], 0-TI-540 Attachment 1, 7/31/2015
Gamma Spectroscopy Results for RCA Release, # 20160227_9 and #20160224_BWH, #20160225_12, #20160201_54, #20160204_14
Personnel Contamination Event Log (PCE #'s 20160001 thru 20160008), 3/1/2016
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Number 16370692, RWCU - HRA, Rev. 0
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Number 16380142, Under Vessel - HRA, Rev. 0
Number 16380404, CRD Exchange - LHRA, Rev. 0
Number 16380407, CRD Exchange - HRA, Rev. 0
Number 16390005, Maintenance Activities - HRA, Rev. 0
RCA Exit Point (LAB) Release Log, 2/4/16 to 3/1/16
Survey Map M-20150914-17, Unit 2 RXB 621' Demin Holding Pump valve packing removal, 9/14/2015
Survey Map M-20150811-4, Unit 2 RXB 621' Demin Holding Pump, 8/11/2015
Survey Map M- 20150818-24, Radwaste 578' & 580', Verification survey due to dose rate alarm, 08/08/2015
TEDE ALARA Evaluations: 16-016, Replace 3-RFV-069-0558, 2/26/16; 16-0035, Cleaning of Bypass Valves with Scotch Brite Pads, 2/26/2016

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CR 1019056
CR 1020064
CR 1023385
CR 1039171
CR 1040530
CR 1042982
CR 1044333
CR 1051741
CR 1067677
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Self-Assessment, BFN-RP-SSA-16-001, Radiation Hazards Analysis and Transportation – October 2015
Quality Assurance Site Audit Report, Audit SSA1506, Radiation Protection Browns Ferry Nuclear Plant August 3 - 14, 2015

Section 2RS8: Radioactive Solid Waste Processing and Radioactive Material Handling
Procedures, Manuals, and Guides
RWI-111, Storage of Radioactive Waste and Materials, Rev. 20
RWTP-101, 10 CFR 61 Waste Characterization, Rev. 2
NPG-SPP-05.9.1, Radioactive Material/Waste Shipments, Rev. 2
0-PCP-001, Process Control Program Manual (PCP), Rev. 4
Certificate of Compliance, 8-120B Type B Cask No. USA/9168/B(U)-96, 8/23/12
NPG-SPP-22.300, Corrective Action Program, Rev. 5

Shipping Records and Radwaste Data
2014 Annual Radioactive Effluent Release Report
Shipping Logs, 3/17/14 – 2/2/16
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Shipment 150820, Type A, Noble Chemistry Coupons
Shipment 150315, Low Specific Activity, Plant Waste
Shipment 160101, Low Specific Activity, Spent Resin
Shipment 141102, Type B, Spent Resin
Shipment 140341, Type B, Filters
Shipment 140102, Type A, Filters
Shipment 140119, Type A, Filters
Shipment 140219, Type A, Filters
Shipment 140301, Type A, Filters
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10 CFR 71.95 Report on Non Conformance Involving Radwaste Cask 8-120B
2014 – 2015 DAW, Analysis of Multiple Data Sets
10 CFR 61 Reactor Water Cleanup Sample Review and Approval, 2014
10 CFR 61 2014 Tri-nuke Filters

CAP Documents
Self-Assessment BFN-RP-SSA-16-001, Radiation Hazards Analysis and Transportation
CR 1118320
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Section 4OA1: Performance Indicator (PI) Verification
Other Documents
NEI 99-02 Regulatory Assessment Performance Indicator Guideline, Rev 7
Unit 1, 2, and 3 Leakage Data
Unit 1, 2, and 3 DEI Data

Section 4OA2: Identification and Resolution of Problems
Other Documents
Control room logs from December 25, 2015
FSAR section 10.14 Control and Service Air, Amendment 26
FSAR section 14.1 Plant Safety Analysis, Amendment 26
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Section 4OA3: Event Follow-up
Procedures
1-OI-32A Unit 1 Drywell Control Air System, Rev 12
2-OI-32A Unit 2 Drywell Control Air System, Rev 43
3-OI-32A Unit 3 Drywell Control Air System, Rev 26

Other Documents
LER 05000259/2015-005-00 Inboard Main Steam Isolation Valve Actuators Inoperable Longer Than Allowed by Technical Specifications
Design Criteria Document for the Containment Inerting System, Rev 18
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Vendor Main Steam Isolation Valve (MSIV) component level analysis dated October 22, 2015
CR 1098857 Insufficient Capability of Inboard MSIV Actuators
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Level 2 Evaluation CR 1073157

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
ADS	Automatic Depressurization System
ARM	area radiation monitor
CAD	containment air dilution
CAP	corrective action program
CCW	condenser circulating water
CFR	Code of Federal Regulations
COC	certificate of compliance
CR	condition report
CRD	control rod drive
CS	core spray
DCN	design change notice
EECW	emergency equipment cooling water
ED	electronic dosimeter
EDG	emergency diesel generator
FE	functional evaluation
FPR	Fire Protection Report
FSAR	Final Safety Analysis Report
HRA	High Radiation Area
HPCI	high pressure coolant injection
IP	Inspection Procedure
IMC	Inspection Manual Chapter
LHRA	Locked High Radiation Area
LER	licensee event report
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSTS	National Source Tracking System
ODCM	Off-Site Dose Calculation Manual
OSLD	Optically Stimulated Luminescence Dosimeter
PCM	Personnel Contamination Monitor
PER	problem evaluation report
PCIV	primary containment isolation valve
PI	performance indicator
PM	Portal Monitor
QA	Quality Assurance
RCA	Radiologically Controlled Area
RCE	Root Cause Evaluation
RCIC	reactor core isolation cooling
RCW	Raw Cooling Water
REMP	Radiological Environmental Monitoring Program
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RPT	Radiation Protection Technician
RS	Radiation Safety

RTP	rated thermal power
RPS	reactor protection system
RWP	radiation work permit
SAM	Small Article Monitor
SDP	significance determination process
SBGT	standby gas treatment
SLC	standby liquid control
SNM	special nuclear material
SR	service request
SRV	safety relief valve
SSC	structure, system, or component
TI	Temporary Instruction
TIP	transverse in-core probe
TRM	Technical Requirements Manual
TS	Technical Specification(s)
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
VHRA	Very High Radiation Area
WO	work order