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

November 14, 2013

EA-13-235

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2013004, 05000260/2013004, AND 05000296/2013004, AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Shea:

On September 30, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. On October 4, 2013, the NRC inspectors discussed the results of this inspection with Mr. K. Polson and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented three findings of very low safety significance (Green) in this report. One of these findings involved a violation of NRC requirements.

If you contest the violations or significance of the non-cited violation (NCV), you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Browns Ferry Nuclear Plant.

In addition, if you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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.

The enclosed report also documents non compliances for which the NRC is exercising enforcement discretion in accordance with Section 9.1 of the NRC Enforcement Policy, “Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48).” The non compliances are associated with your implementation of the requirements and standards of your technical specifications, as well as 10 CFR 50, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.” The inspectors have screened the violations, identified in sections 4OA3.3-7 of this report, and determined that they warrant enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues, and Inspection Manual Chapter 0305, Operating Reactor Assessment Program, Section 11.05.b.

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 the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/William Jones RA for/

Richard P. Croteau, Director
Division of Reactor Projects

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

Enclosure: NRC Integrated Inspection Report 05000259/2013004,
05000260/2013004 and 05000296/2013004

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the violation and determined that it warrants enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues, and Section 11.05.b of IMC 0305.

In accordance with 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 the NRC’s Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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Letter to Joseph W. Shea from Scott Shaeffer dated November 14, 2013.

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2013004, 05000260/2013004, AND 05000296/2013004, AND EXERCISE OF ENFORCEMENT DISCRETION

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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.:	05000259/2013004, 05000260/2013004, 05000296/2013004
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:	July 1, 2013, through September 30, 2013
Inspectors:	D. Dumbacher, Senior Resident Inspector L. Pressley, Resident Inspector T. Stephen, Resident Inspector M. Riches, Resident Inspector (Acting) J. Hamman Resident Inspector (Acting) D. Jones, Senior Reactor Inspector J. Montgomery, Reactor Inspector
Approved by:	Scott Shaeffer, Chief Reactor Projects Special Branch Division of Reactor Projects

SUMMARY

IR 05000259/2013004, 05000260/2013004, 05000296/2013004; 04/01/2013–06/30/2013; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Maintenance Risk Assessment and Emergent Work Evaluation, Operability Evaluations, and Follow-up of Events and Notices of Enforcement Discretion.

The report covered a three month period of inspection by the resident inspectors. The significance of most findings is identified by their color (Green, White, Yellow, and Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); and, the cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas". Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 4, dated December 2006.

NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The NRC identified a Green finding for the licensee's failure to properly screen and classify corrective action program (CAP) problem evaluation reports (PER's) in accordance with NPG-SPP-03.1.4, Corrective Action Program Screening and Oversight. Specifically, the licensee failed to screen service requests (SR's) that had a high potential for resulting in a reactor scram as 'A' level PER's. The licensee entered the issue into the corrective action program as PER 687732.

  This finding was determined to be more than minor because, if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The finding was associated with the Initiating Events cornerstone and using IMC 0609, Appendix A, At-Power Significance Determination Process screening questions for transient initiators, the finding screened as Green because the finding did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The cause of this finding was directly related to the cross-cutting aspect of thoroughly evaluating problems such that the resolutions address causes and extent of condition in the corrective action program component of Problem Identification and Resolution. [P.1.c] (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. The NRC identified a non-cited violation (NCV) of Technical Specifications (TS) 5.4.1.a, Procedures, for the licensee's failure to follow OPDP-8, Operability Determination Process and Limiting Conditions for Operation Tracking. Specifically, the licensee failed to enter a seven day action statement C.1 of Technical Specification 3.7.1, Residual Heat Removal Service Water (RHRSW) system and Ultimate Heat Sink when planned maintenance rendered two RHRSW pumps inoperable. The licensee entered this issue into their corrective action program as Problem Event Report (PER) 751300.

This finding was determined to be more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The finding affected the Mitigating Systems cornerstone and using IMC 0609.04, Initial Characterization of Findings and IMC 0609 Appendix A, Exhibit 2 Mitigating Systems screening questions, the finding screened as very low safety significance (Green). The finding did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time and did not represent an actual loss of function of one or more non-technical specification equipment for greater than 24 hours because the licensee restored the C1 and C2 RHRSW pumps on July 5, 2013. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance associated with the work practices component because the licensee failed to ensure that expectations for procedural compliance were properly communicated and that personnel followed procedures. [H.4.b]. (Section 1R13)

- Green. An NRC identified finding (FIN) was identified for the licensee's exceeding the maximum allowed periodicity for inspecting and cleaning the Residual Heat Removal Service Water (RHRSW) pump pit per Raw Water Corrosion Program procedure (NPG-SPP 9.7.3).

  This finding was determined to be more than minor because, if left uncorrected, the failure to maintain the intake pump pit cleaning would have had the potential to lead to a more significant safety concern in that, it could lead to fouling of safety related coolers. challenging the heat exchanger heat removal function. The finding is associated with the Mitigating Systems cornerstone. Using IMC 0609 Appendix A, Exhibit 2, the finding screened as green because the finding did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time and did not represent an actual loss of function of one or more non-technical specification equipment for greater than 24 hours. The cause of this finding was associated with the human performance area, resources component, cross cutting aspect of maintaining long term plant safety by maintenance of design margins and minimizing preventative maintenance deferrals due to the licensee not allocating resources to clean the intake pump pits. [H.2.(a)]. (Section 1R15)

Licensee Identified Violations

No findings were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at 100 percent of rated thermal power (RTP) except for three planned downpowers, July 20, 2013, for 1A2 feedwater heater repair, July 23, 2013, for 1C reactor feedwater pump repairs and September 8, 2013, for replacement of 1B reactor feedwater pump 24VDC power supply. On September 23, 2013, an unplanned power reduction to 50 percent occurred for repairs to 1C1 moisture separator high level dump valve. Unit 1 remained at RTP for the remainder of the quarter.

Unit 2 operated at 100 percent of rated thermal power (RTP) Except for two planned downpowers; July 9, 2013, for 2A reactor feedwater pump and 2B2 feedwater heater level control valve repairs and September 6, 2013, waterbox cleaning, and 2A condensate booster pump leak repairs. On August 15, 2013, an unplanned downpower to 45 percent occurred to repair '2B' reactor feedwater pump redundant power supply. Power remained at RTP for the remainder of the quarter.

Unit 3 operated at 100 percent of rated thermal power (RTP) except for one planned downpower for rod sequence exchange on August 31, 2013, and two unplanned downpowers; July 29, 2013, to maintain condenser vacuum and September 8, 2013, to perform EHC oil leak repairs. Power remained at RTP for the remainder of the quarter.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

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

- July 16, 2013, Standby Gas Treatment System Train C
- September 24, 2013, Core Spray System Unit 1 Division I

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 3 RHR Division II System, using the applicable P&ID flow diagrams, 3-47E811-1 and 3-47E858-1, along with operating instruction, 3-OI-74, to verify equipment availability and operability. The inspectors reviewed relevant portions of the UFSAR and technical specifications. This detailed walkdown also verified electrical power alignment, the condition of applicable system instrumentation and controls, component labeling, pipe hangers and support installation, and associated support systems status. Furthermore, the inspectors examined applicable Operator Workarounds (OWAs), open Work Orders (WOs), and any previous Problem Evaluation Reports (PERs) that could affect system alignment and operability. This activity constituted one Complete Walkdown inspection sample.

- July 30, 2013, Unit 3 Residual Heat Removal System, Division II

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Fire Protection Tours

a. Inspection Scope

The inspectors reviewed licensee procedures, Nuclear Power Group Standard Programs and Processes (NPG-SPP)-18.4.7, Control of Transient Combustibles, and NPG-SPP-18.4.6, Control of Fire Protection Impairments, and conducted a walkdown of five fire areas (FA) and fire zones (FZ) listed below. Selected FAs/FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure NPG-SPP-18.4.6. Furthermore, the inspectors reviewed applicable portions of the Fire Protection Report, Volumes 1 and 2, including the applicable Fire Hazards Analysis, and Pre-Fire Plan drawings, to verify that the necessary firefighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, was in place. This activity constituted six Fire Protection inspection samples.

- July 7, 2013, Unit 1 Reactor Building, EL 593’, North of Column R (Fire Area 1-3)
- July 7, 2013, Unit 1 Reactor Building, EL 593’, South of Column R (Fire Area 1-4)
- August 7, 2013, Unit 1 Reactor Building, Elevations 621’ and 639’ north of column line R (Fire Zone 1-5)
- August 9, 2013, Unit 1 and 2 Emergency Diesel Generator rooms (Fire Area 20)

- September 9, 2013, Intake Pumping Station (Fire Area 25-1)
- September 9, 2013, Intake Pumping Station (Fire Area 25-2)

b. Findings

No findings were identified.

1R06 Flood Protection Measures

.1 Cables Located in Underground Bunkers/Manholes

a. Inspection Scope

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

The inspectors reviewed past preventative maintenance activities performed by the licensee to inspect plant manholes, valve pits, and cable tunnels; and check operability of applicable sump pumps. The inspectors reviewed the potential effects of a design basis flood on the Handhole 15 and Manhole 26 cables. This activity constituted one Underground Manhole flooding inspection sample.

b. Findings

No findings were identified.

.2 Internal Flooding

a. Inspection Scope

The inspectors performed a walkdown of the Unit 1, 2, and 3 Emergency Diesel Generator (EDG) rooms to review flood-significant features of floor drains that might be subjected to flood conditions. Specifically, the inspectors reviewed the design basis pipe crack scenario as required by Browns Ferry Civil Design Criteria BFN-50-C-7105 and NRC Standard Review Plan 3.6.1

The inspectors also reviewed a sampling of the licensee's corrective action documents with respect to flood-related items to verify that problems such as clogged drains were being identified and corrected. The inspectors reviewed selected completed preventive maintenance procedures, work orders, and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents. This activity constituted one Flood Protection measures inspection sample.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification and Performance

.1 Licensed Operator Requalification

a. Inspection Scope

On August 12, 2013, the inspectors observed an as-found licensed operator requalification for an operating crew according to Unit 2 Simulator Exercise Guide (SEG) OPL177.108, MSIV Partial Close, and Safe Shutdown Instruction (SSI) Fire.

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 administer testing and assess the performance of their licensed operators. The inspectors attended the post-examination critique performed by the licensee evaluators, and verified that licensee-identified issues were comparable to issues identified by the inspector. The inspectors also reviewed simulator physical fidelity (i.e., the degree of similarity between the simulator and the reference plant control room, such as physical location of panels, equipment, instruments, controls, labels, and related form and function). This activity counts for one Observation of Requalification Activity inspection sample.

b. Findings

No findings were identified.

.2 Control Room Observations

a. Inspection Scope

Inspectors observed and assessed licensed operator performance in the plant and main control room, particularly during periods of heightened activity or risk and where the activities could affect plant safety. Inspectors reviewed various licensee policies and procedures such as OPDP-1, Conduct of Operations, NPG-SPP-10.0, Plant Operations and GOI-100-12, Power Maneuvering.

Inspectors utilized activities such as post maintenance testing, surveillance testing and refueling and other outage activities to focus on the following conduct of operations as appropriate;

- Operator compliance and use of procedures.
- Control board manipulations.
- Communication between crew members.
- Use and interpretation of plant instruments, indications and alarms.
- Use of human error prevention techniques.
- Documentation of activities, including initials and sign-offs in procedures.
- Supervision of activities, including risk and reactivity management.
- Pre-job briefs.

This activity constituted one Control Room Observation inspection sample.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

.1 Routine

a. Inspection Scope

The inspectors reviewed 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 (i.e., Ten Point Plan). The inspectors also compared the licensee's performance against site procedure NPG-SPP-03.4, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; and NPG-SPP-03.1/NPG-SPP-22.300, Corrective Action Program. The inspectors also reviewed, as applicable, work orders, surveillance records, PERs, system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met. This activity constituted four Maintenance Effectiveness inspection samples.

- Instrument Line Excess Flow Check Valve Maintenance, (Unit 2)
- 3C emergency diesel not producing an adequate field flash.
- Safety related doors (System 303) ability to perform their design basis function.
- B3 Emergency equipment cooling water pump failure to start during testing.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

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

- July 1, 2013; "C" RHRSW header outage with Main Battery Bank #1 out of service
- July 24, 2013, Unit 1 and 2 "D" Emergency Diesel Generator out-of-service (OOS) for extended diesel generator outage, with B3 and D3 emergency equipment cooling water (EECW) pumps unavailable
- August 13, 2013; EDG 3D, 3EN LPCI MG Set, B3,C3, D1 RHRSW/EECW, and 3D RHR pumps all out of service
- August 26, 2013, During Division I work week, 4Kv shutdown board C (Division II) normal feeder breaker 1718 opened during parallel evolution with inoperable C EDG
- September 16, 2013 'C' RHRSW header and Unit 2 HPCI out of service, 161 kV grid status yellow
- September 30, 2013, Main Bank One (MB-1) battery unplanned out of service.

a. Findings

.1 Introduction: The NRC identified a Green NCV of TS 5.4.1.a, Procedures, for the licensee's failure to implement OPDP-8, Operability Determinations and Limiting Conditions for Operations (LCO) tracking. Specifically, the licensee failed to track the applicability of action statement C.1 of TS LCO 3.7.1, Residual Heat Removal Service Water (RHRSW) system and Ultimate Heat Sink, during planned maintenance.

Description: From July 1, 2013, until July 11, 2013, the licensee performed piping replacements on the "C" RHRSW header as part of planned maintenance activities. During this period, all three units were fueled, therefore, four RHRSW subsystems and eight RHRSW pumps were required to be operable in accordance with TS 3.7.1 Residual Heat Removal Service Water (RHRSW) system and Ultimate Heat Sink. At the start of this maintenance the C1 and C2 RHRSW pumps had their breakers racked out to prevent inadvertent pump operation. Procedure OPDP-8, "Operability Determination Process and Limiting Conditions for Operation Tracking," Revision 16, section 3.1, required, in part, plant log entries of entry and exit from technical specification action statements. The licensee appropriately entered TS LCO required action 3.7.1.B.1 due to one RHRSW subsystem being inoperable for all 3 units which required the subsystem to be returned to service in 30 days. In addition to this requirement, LCO 3.7.1.C.1 required 1 RHRSW pump to be restored to operability in 7 days should 2 RHRSW pumps be declared inoperable. When the NRC inspector challenged the licensee on the applicability of TS LCO 3.7.1.C.1, the licensee generated a position paper detailing their interpretation of the TS 3.7.1 conditions in which they determined the LCO did not apply. As part of the maintenance, a modification to the piping was completed on July 5, 2013, which allowed the C1 and C2 RHRSW pumps to be restored to operation to support Units 1 and 3. On July 8, 2013, the licensee reanalyzed the situation and concurred with the NRC's assessment of their non-compliance and documented this issue and their corrective actions in PER 751300.

Analysis: The licensee's failure to track applicable technical specification action statements as required by section 3.1 of OPDP-8, "Operability Determination Process and Limiting Conditions for Operation Tracking" was a performance deficiency. This finding was determined to be more than minor because, if left uncorrected, it would have had the potential to lead to a more significant safety concern in that, the failure to track an applicable technical specification action statement could lead to plant operations outside of TS analyzed conditions. Using MC 0609.04, "Initial Characterization of Findings," and MC 0609 Appendix A, "Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, the finding screened as very low safety significance (Green) because it did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time and did not represent an actual loss of function of one or more non-technical specification equipment for greater than 24 hours. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance associated with the work practices component because the licensee failed to ensure that expectations for procedural compliance were properly communicated and personnel follow procedures. Specifically, technical specification conditions for the RHRSW pumps were not followed. [H.4.b].

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 16. OPDP-8, section 3.1, required, in part, plant 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 3.7.1 condition C.1 from July 1-5, 2013. The required RHRSW pumps were restored to operable status on July 5, 2013. Because this finding was entered into the licensee’s corrective action program as PER 751300, it is being treated as an NCV, consistent with section 2.3.2 of the Enforcement Policy and is identified as NCV 05000259, 260, 296/2013004-01, Failure to Track Applicable Technical Specification Action Statement for Residual Heat Removal Service Water System.

1R15 Operability Determinations and Functionality Assessment

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 also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedure NEDP-22, Functional Evaluations, and NEDP-27, Past Operability Evaluations, to ensure that the licensee’s evaluation met procedure requirements. 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. This activity constituted eight Operability Evaluation inspection samples.

- Unit 2 Residual Heat Removal (RHR) heat exchanger 2A past operability due to excessive Asiatic clams (PER 732555)
- Unit 1 High Pressure Coolant Injection (HPCI) Main Pump Seal Leakage (PER 734876)
- Unit 1/2 ‘D’ Emergency Diesel Generator Operability Impacts Associated With Unexpected Shutdown During Performance of WO 114863764, Governor Setup and Tuning Instruction (PER 758756)
- ‘D’ / ‘3D’ Emergency Diesel Generator Frequency Lower than Expected during Parallel Testing, (PER 767182)
- Unit 1 A RHR minimum flow valve stroked closed early due to system voiding, (PER 763930)
- Unit 1/2 ‘C’ Emergency Diesel Generator battery high resistance (PER 766439)

- Review of 1B Standby Liquid Control pump trip (PER 681667)
- Unit 1/2 'C' Emergency Diesel Generator skid hold down bolt would not tighten, (PER 768716)

b. Findings

Introduction: The NRC identified a Green finding for the licensee's exceeding the maximum allowed periodicity for inspecting the Residual Heat Removal Service Water (RHRSW) pump pit per Corrosion Control Procedure (SPP-9.7) and Raw Water Corrosion Program procedure (NPG-SPP-09.7.3).

Description: Between April 2013 and September 2013, the NRC inspected a combination of six RHR and diesel generator heat exchangers that the licensee had opened for scheduled cleaning. The NRC had identified that the first two inspections had not met the acceptance criteria and had not been performed within the scheduled frequency. Additionally the NRC identified that the established method (prior to 2013) for operability was not accounting for clam fouling. Five of the six inspected heat exchangers exceeded the licensee's acceptance criteria for fouling. In one case, the results were in excess of 300 percent of the acceptance criteria. After each discovery of excessive fouling, the licensee performed a significantly more detailed operability analysis and concluded the heat exchangers could still remove sufficient heat energy.

NRC inspectors challenged the initial root cause of the heat exchangers excessive fouling and the failure to perform procedural based inspections on time. The licensee changed the root cause to be related to the pump pit and moved up the inspection of the RHRSW pump pit to September 2013. Several piles of clam shells and silt, as high as six feet, were found in the pump pit near each of the service water pumps that serve the RHR and EDG heat exchangers. The licensee causal analysis determined the final root cause to be insufficiently supported deferrals of the pump pit cleanings.

In letters dated March 16, 1990, and November 8, 1999, TVA committed to aspects of NRC Generic Letter (GL) 89-13. At Browns Ferry these commitments included inspecting and cleaning the raw water side of the Residual Heat Removal (RHR) heat exchangers (HX), inspecting the Residual Heat Removal Service Water (RHRSW) pump pit, and inspecting and cleaning the Emergency Diesel Generator (EDG) HX's. Browns Ferry has previously experienced a long term historical Asiatic clam fouling.

The licensee's inspection and cleaning requirement for the RHRSW pump pit was at least once every cycle (24 months) unless a technical justification can be made to increase the inspection frequency to once per two cycles. These requirements are captured in Corrosion Control Procedure (SPP-9.7) and Raw Water Corrosion Program Procedure (NPG-SPP-9.7.3) to ensure commitments to the NRC are met. No allowance is made for deferrals beyond once per two cycles. The maximum procedural allowed time between inspections was exceeded three times; once between 1998 and 2003, once between 2003 and 2008, and once between 2008 and 2013.

Analysis: The failure to inspect the Residual Heat Removal Service Water pump pit once per two cycles or less in accordance with SPP-9.7 Corrosion Control Procedure and NPG-SPP 9.7.3 Raw Water Corrosion Program was a performance deficiency. This finding was determined to be more than minor because, if left uncorrected, the failure to maintain the intake pump pit cleaning would have had the potential to lead to a more significant safety concern in that, it could lead to fouling of safety related coolers challenging the heat exchanger heat removal function. The finding is associated with the Mitigating Systems cornerstone. Using Inspection Manual Chapter (IMC) 0609 Appendix A, Exhibit 2, the finding screened as green because the Residual Heat Removal (RHR) and Diesel heat exchangers were determined to maintain operability after additional analysis. The licensee's immediate corrective action was to clean all heat exchangers and reschedule to an earlier date cleaning of the RHR service water pump pit. The cause of this finding was associated with the human performance area, resources component, cross cutting aspect of maintaining long term plant safety by maintenance of design margins and minimizing preventative maintenance deferrals due to the licensee not allocating resources to clean the intake pump pits [H.2.(a)].

Enforcement: This finding does not involve enforcement action because no regulatory requirement violation was identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as a FIN [05000259/260/296/2013004-02], "Failure to Clean the Safety Related Pump Pit Once per Two Cycles." This Green finding closes out the Unresolved Item (URI) 05000260/2013003-03 from Browns Ferry Inspection Report number 2013003.

1R18 Plant Modifications

.1 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the maintenance and replacement of the Unit common RHRSW room doors at the intake pump house (DCN 70735A). The inspectors reviewed licensee procedures NPG-SPP-09.3, Plant Modifications and Engineering Change Control, NPG-SPP-09.4, 10 CFR 50.59 Evaluations of Changes, Tests, and Experiments, NPG-SPP-09.20, Vendor Manual Control and NPG-SPP-06.9.3, Post-Modification Testing, and observed part of the licensee's activities to implement this modification. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify that the modifications had not affected system operability/ availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents. This activity constituted one Permanent Plant Modification sample.

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors witnessed and reviewed the seven 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, and that the procedure had been properly reviewed and approved. The inspectors also witnessed and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors verified that PMT activities were conducted in accordance with applicable WO instructions, or licensee procedural requirements. Furthermore, the inspectors verified that problems associated with PMTs were identified and entered into the CAP. This activity constituted seven Post Maintenance Test inspection samples.

- July 3, 2013, 250V Battery charger #1 for Main Bank Battery #1 (WO 114844701)
- July 25, 2013, Unit 1 and 2 Diesel Generator D Governor Setup and Tuning (WO 114863764)
- August 14, 2013, PMT 'D' / '3D' Emergency Diesel Generator Units in Parallel Mode of Operation per PMTI-69532-STG004
- August 14, 2013, Unit 1: PMT for 1B Standby Liquid Control Pump per 1-SI-4.4.A.1, Standby Liquid Control Pump Functional Test (WO 114422086)
- August 31, 2013, Unit 3: Scram time testing for Rod 42-15 following air leak repair on Hydraulic Control Unit (WO 114087239)
- September 3, 2013, Unit 3: Verification of fuel rack positions following a full load run as part of a PMT for '3B' Emergency Diesel Generator (WO 114395313)
- September 20, 2013, Standby Gas Treatment Train C post maintenance test (WO 114500098)

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions of, and/or reviewed completed test data for the following surveillance tests of risk-significant and/or safety-related systems to verify that the tests met technical specification surveillance requirements, UFSAR commitments, and in-service testing and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the SSCs were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement. This activity constituted eight inspection samples: one in-service and seven routine tests.

<u>In-Service Tests</u>:

- August, 1, 2013, 3-SR-3.5.3.3(COMP), RCIC Comprehensive Pump Test

<u>Routine Surveillance Tests</u>:

- July 8, 2013,0-SR-3.6.4.1.3, Combined Zone Secondary Containment Drawdown and Integrity Test, Rev. 0018
- July 26, 2013, Core Spray Flow Rate Loop II (3-SR-3.5.1.6(CS II))
- July 29, 2013, Routine Diesel Generator A Monthly Operability Test
- August 20, 2013, 0-TI-19, Reactor Vessel Fatigue Usage Factor Evaluation, Monitoring, Recording, Evaluating, and Reporting
- September 4, 2013, 1-SR-3.5.3.3 – RCIC System Rated Flow at Normal Operating Pressure (WO 114489624)
- September 20, 2013, 0-SR-3.6.4.3.2- C SGT Filter Train Humidity Control Heater Test
- September 20, 2013, 2-SR-3.5.1.7- HPCI System Rated Flow Test

b. <u>Findings</u>

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 <u>Drill Evaluation</u>

a. <u>Inspection Scope</u>

During the report period, the inspectors observed an Emergency Preparedness (EP) drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures on August 7, 2013, to identify any weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room and certain Emergency Response Facilities to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure and other applicable Emergency Plan Implementing Procedures. The inspectors also attended the post-drill critique to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying weaknesses. This inspection activity satisfied one inspection sample for the Drill Evaluation of emergency preparedness.

b. <u>Findings</u>

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Mitigating Systems

Mitigating Systems

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following Performance Indicators (PIs), including procedure NPG-SPP-02.2 Performance Indicator Program. The inspectors examined the licensee's PI data for the specific PIs listed below for the third quarter 2012 through second quarter of 2013. The inspectors reviewed the licensee's data and graphical representations as reported to the NRC to verify that the data was correctly reported. The inspectors also validated this data against relevant licensee records (e.g., PERs, Daily Operator Logs, Plan of the Day, Licensee Event Reports, etc.), and assessed any reported problems regarding implementation of the PI program. Furthermore, the inspectors met with responsible plant personnel to discuss and go over licensee records to verify that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors also used the Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to ensure that industry reporting guidelines were appropriately applied. This activity constituted twelve mitigating systems performance indicator inspection samples.

- Unit 1 Mitigating Systems Performance Index – Safety System Functional Failures
- Unit 2 Mitigating Systems Performance Index – Safety System Functional Failures
- Unit 3 Mitigating Systems Performance Index – Safety System Functional Failures
- Unit 1 Mitigating Systems Performance Index – Cooling Water System
- Unit 2 Mitigating Systems Performance Index – Cooling Water System
- Unit 3 Mitigating Systems Performance Index – Cooling Water System
- Unit 1 Mitigating Systems Performance Index – Emergency Alternating Current
- Unit 2 Mitigating Systems Performance Index – Emergency Alternating Current
- Unit 3 Mitigating Systems Performance Index – Emergency Alternating Current
- Unit 1 Mitigating Systems Performance Index – Residual Heat Removal
- Unit 2 Mitigating Systems Performance Index – Residual Heat Removal
- Unit 3 Mitigating Systems Performance Index – Residual Heat Removal

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution of Problems

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

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

.2 Focused Annual Sample Review - Operator Workarounds

a. Inspection Scope

The inspectors conducted a review of existing OWA to verify that the licensee was identifying OWAs at an appropriate threshold, entering them into the corrective action program, establishing adequate compensatory measures, prioritizing resolution of the problem, and implementing appropriate corrective actions in a timely manner commensurate with its safety significance. At the time of the inspection a total of 87 OWAs were active. The inspectors examined all active OWAs listed in the Limiting Condition of Operation Tracking (LCOTR) Log, and reviewed them against the guidance in BFN-ODM-4.16, Operator Workarounds/Burdens/Challenges. The inspectors conducted a sampling of all OWAs for selected positions to verify the cumulative burden for any one position did not exceed a threshold value of 60 minutes in additional time added to the position's normal routine activities. The inspector also reviewed the licensee's latest self-assessment of OWAs. The inspectors performed a focused inspection on all of the Priority 1 OWAs. The Priority 1 OWAs address compensatory actions that must be implemented during abnormal or emergency operations to compensate for equipment or program deficiencies. This focused review for selected OWAs included the applicable PER, including any associated functional evaluation and corrective action plan (both interim and long term). This activity constituted one Operator Workaround sample.

b. Assessment and Observations

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

Inspectors determined that Browns Ferry adequately tracks and trends all operator workarounds, burdens and challenges. This included estimating, tracking and compiling the aggregate impact of the workarounds, burdens and challenges. Inspectors determined that the licensee routinely performed self-assessments of the OWAs. However, the self-assessment focused mostly on determining the aggregate burden of the OWAs for a given position. It did not consider other aspects of the OWAs such as feasibility of the actions or length of time the OWA had been active. Thirteen Priority 1 OWAs were active. Of these, the inspectors identified issues with twelve of the thirteen Priority 1 OWAs. Most of the issues were with the length of time the OWA has been or will be active and the feasibility of some of the actions. The OWAs were not immediately available to the operating crews. The emergency/abnormal procedures these OWAs affected had no cue for the user that an OWA applied. Successful performance of the actions in OWAs resided in the “knowledge-based” realm rather than “procedure-driven” realm of operations.

.3 Focused Annual Sample Review – Service Water Clam Fouling

a. Inspection Scope

The inspectors conducted a review of challenges to operability of service water cooled heat exchangers to verify that the licensee was identifying non-conformances, entering them into the corrective action program, establishing adequate compensatory measures, prioritizing resolution of the problem, and implementing appropriate corrective actions in a timely manner commensurate with its safety significance. During the period of April through September 2013, six safety related heat exchangers that fall under the Generic Letter 89-13 “Service Water System Problems Affecting Safety-Related Equipment” program were opened for inspection. Of the six heat exchangers, four had Asiatic clam shell fouling that exceeded acceptance criteria. The heat exchangers included both the Residual Heat Removal system and the Emergency Diesel Generators. The clam fouling challenged the ability of these heat exchangers to perform their design basis functions. The licensee’s analysis of the causes and corrective actions were inspected. As a result, increased analysis of this issue was conducted by the licensee culminating in a series of corrective actions designed to minimize future occurrences of clam fouling.

b. Assessment and Observations

One finding was identified and is captured in section 1R15. Additionally, the inspectors had the following observations which were discussed with the licensee:

The frequency and adequacy of the cleaning of the service water pump pit was not adequate to minimize challenges to operability of the heat exchangers it serviced. The pump pit received its last full cleaning in 2008 and received a partial cleaning in 2009. The frequency of the inspection and cleanings of the Residual Heat Removal heat exchangers did not occur frequently enough to minimize challenges to their acceptance criteria. The licensee was unable to produce any analysis that supported a shift in RHR heat exchanger cleaning frequency from once every 18 months to once every 48 months. The practice of not performing flow rate checks for the Emergency Diesel Generator heat exchangers while they were in a protected equipment status was not considered conservative. This practice prevented the timely discovery of challenges to the heat exchanger due to fouling. This was especially pronounced during extended Emergency Diesel Generator repair periods where periods of up to 14 days passed without a flow rate check.

4OA3 Follow-up of Events and Notices of Enforcement Discretion

.1 (Closed) Licensee Event Report (LER) 05000259/2013-002-00, Manual Reactor Shutdown Due to Decreasing Condenser Vacuum

a. Inspection Scope

On March 19, 2013, at approximately 0402 hours Unit 1 was manually scrammed due to decreasing main condenser vacuum. The degrading condenser vacuum was caused by the failure of a vent and drain pipe joint to a miscellaneous drain header which connected to the main condenser. The licensee initiated PER 698870 to enter the event into the corrective action program. Inspectors reviewed licensee event report 05000259/2013-002-00 and all associated documentation which included the Root Cause Analysis (RCA) for PER 698870. The direct cause of the event was attributed to cyclic fatigue of the drain piping to header connection. The root cause was attributed to the valve leakage and the station's failure to consider risk associated with the possible failure of the piping.

This LER is closed.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000296/2013-003-00, Automatic Reactor Shutdown Due to an Actuation of the Reactor Protection System from a Turbine Trip.

a. Inspection Scope

On February 25, 2013, at 1313 hours, Unit 3 automatically scrammed due to a turbine trip. The turbine trip was caused by a loss of condenser vacuum as a result of a failure of a long cycle return line piping connection to the miscellaneous drain header. Inspectors reviewed 05000296/2013-003-00 and all associated documentation which included the Root Cause Analysis (RCA) for PER 687732.

This LER is closed.

b. Findings

Introduction: The NRC identified a Green finding for the licensee's failure to properly screen and classify corrective action program (CAP) problem evaluation reports (PER's) in accordance with NPG-SPP-03.1.4, Corrective Action Program Screening and Oversight. Specifically, the licensee failed to screen service requests (SR's) that had a high potential for resulting in a reactor scram as 'A' level PER's.

Description: During review of the licensee's actions following a loss of condenser vacuum that was caused by a failure of the feedwater long cycle return line piping, the inspectors questioned the screening of previous corrective action items associated with this system. On August 23, 2012, PER 600143 was initiated for a piping leak downstream of the Unit 3 long cycle valves. This PER was screened as a ‘D’ level or low level, low priority issue. WO 113809831, initiated from PER 600143, later repaired the leak and the problem was considered resolved.

PER 600927 was initiated as a QA intervention regarding the risk decision making associated with the piping leaks. This PER detailed the continued degradation of the pipe leak and piping vibration and concluded that the station was not actively engaged in evaluating the condition. Based upon a known history of issues, and experience, concerning these valves and associated piping the licensee should have concluded that a failure of the 8 inch long cycle piping could result in a potential significant condenser vacuum issue. Given the QA intervention from PER 600927, which identified a degradation of the system, the licensee should have rescreened PER 600143 as an ‘A’ level PER according to NPG-SPP-03.1.4, Corrective Action Program Screening and Oversight, Appendix A, because the procedure required an ‘A’ level screening result for any unplanned plant operating condition or equipment failure with a high potential for resulting in a reactor scram.

Analysis: The licensee’s failure to properly screen and classify corrective action program PER’s that had a high potential for resulting in a reactor scram according to NPG-SPP-03.1.4, Corrective Action Program Screening and Oversight, Appendix A, was a performance deficiency. The inspectors determined this finding was more than minor because if left uncorrected it had the potential to lead to a more significant safety concern. Specifically, failure to identify and correct the root causes of the piping leaks would potentially lead to additional, potentially risk significant, reactor scrams.

The inspectors determined this finding was associated with the Initiating Events cornerstone and evaluated the significance using IMC 0609, Appendix A, for At-Power Significance Determination Process. The issue screened as Green using the Initiating Events Screening Questions for transient initiators because the finding did not cause both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The cause of this finding was directly related to the cross-cutting aspect of thorough evaluation of problems, in the corrective action program component, of the problem identification and resolution area. Specifically, the licensee failed to thoroughly evaluate long cycle piping leaks to allow for proper classification of the issues in the corrective action program. [P.1(c)]

Enforcement: This finding does not involve enforcement action because no regulatory requirement violation was identified. This issue was entered into the licensee’s corrective action program as PER 687732. This finding contributes to the closure of Unresolved Item (URI) 05000259, 260, 296/2013003-02 opened in Browns Ferry report 2013003. Because this finding does not involve a violation and is of very low safety significance it is identified as a FIN 05000296/2013004-04, Failure to Properly Screen and Classify Corrective Action Program, Problem Evaluation Reports.

3 (Closed) Licensee Event Report (LER) 05000259, 260, 296/2009-005-00, 01, and 02: Reactor Vessel Water Level 1 Initiation Logic Including the Common Accident Logic Not Evaluated for Appendix R Fire Event

a. Inspection Scope

On October 20, 2009, the licensee submitted an LER documenting the discovery of a condition of non-compliance with the Appendix R Safe Shutdown Program. This condition could prevent operators from achieving and maintaining safe shutdown (SSD) of the plant, in the case of a postulated fire.

The inspectors performed a detailed review of the information related to the LER. Inspectors reviewed documents, performed walkdowns, and discussed the event with plant personnel to gain an understanding of the event. The inspectors assessed the licensee's compensatory measures and corrective actions to determine if they were adequate.

The LERs are closed.

b. Findings

Introduction. A licensee identified non-compliance with TS 5.4.1.a. was identified for inadequate procedural guidance. Specifically, the licensee failed to consider the automatic response of plant control systems to predicted plant parameters in the Appendix R Safe Shutdown Analysis (SSA). As a result, SSD procedures did not include steps that may have been required to establish needed system alignments.

Description. On October 20, 2009, the licensee submitted LER 2009-005-00, describing a condition that could have prevented the fulfillment of the safety function of structures or systems needed to shutdown the reactor and maintain it in a safe shutdown condition. Subsequently, the licensee performed engineering evaluations and sensitivity evaluations, and based on the results, submitted revisions to the LER on November 3, 2009, and August 31, 2010.

While performing an evaluation of the simulator response during a training exercise, site engineering personnel identified a condition where, for a plant shutdown requiring 20 minute depressurization, reactor vessel water level could momentarily drop below the Reactor Vessel Water Level 1 setpoint (-122 inches). Dropping below this setpoint could result in automatic realignment of several SSD systems. The licensee determined that this condition was applicable to 11 fire areas.

For a postulated fire in certain areas, SSD of the plant is predicated upon the availability of certain equipment. The plant's Appendix R SSA identifies the SSD strategy for fires in certain areas, and the equipment available to implement the strategy. For a postulated fire that requires a plant shutdown with 20 minute depressurization, dropping below the Reactor Vessel Water Level 1 setpoint could cause automatic realignment of SSCs that are credited in the plant's Appendix R SSA. This would require operators to reestablish Appendix R system alignments to achieve and maintain SSD. The BFN Safe Shutdown

Methodology, established in 1986, did not consider the impacts of the Level 1 logic initiation. Thus, the actions necessary to reestablish the needed system alignments were not specified in SSD procedures. For postulated fires, failure to maintain the required Appendix R system alignments could lead to fuel cladding temperature exceeding 1,500°F, which could cause fuel damage. The requirement for fuel cladding to remain below 1,500°F is specified in the site's fire protection report as a SSD system requirement.

Upon discovery, the licensee entered the condition into the corrective action program (PER 177130), and implemented compensatory measures. These compensatory measures included operator manual actions (OMAs) to inhibit the accident logic after safe shutdown instruction (SSI) entry. The licensee also added an additional Auxiliary Unit Operator to the minimum shift staffing requirements to support the additional OMAs. The licensee also implemented fire watches as additional compensatory measures.

Analysis. The licensee's failure to meet TS requirements due to inadequate procedural guidance is a performance deficiency. This finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and it negatively affected the cornerstone objective of protection against external events such as fire to prevent undesirable consequences. Because this issue relates to fire protection, and this non-compliance was identified as a part of the site's transition to NFPA 805, this issue is being dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy.

In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), a bounding phase 3 SDP risk analysis was performed by a regional SRA using the guidance from NRC IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005, and NUREG/CR 6850 Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," dated September 2010. For each SSI where the potential existed to actuate the Reactor Vessel Water level 1, the bounding analysis assumed a conditional core damage probability of 1.0 for all the credible fire sequences. The credible fire sequences used were those developed for the BFNP Operator Manual Action SDP Risk Analysis (EA 09-307). Based on this bounding risk analysis, the regional SRA determined that this performance deficiency resulted in a CDF increase for each BFNP unit of less than 1E-4/year (i.e., less than Red).

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

Enforcement. TS 5.4.1.a. requires that written procedures shall be established, implemented, and maintained covering the activities in NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2. Regulatory Guide 1.33, Appendix A, Section 6.v, requires procedures for combating emergencies such as plant fires. Embodied within these requirements is the requirement that the procedures are adequate.

Contrary to the above, procedural guidance given in SSIs 1-1, 2-1, 2-2, 2-3, 2-4, 3-1, 3-2, 3-3, 9, 12, and 13 was inadequate. Specifically, on October 20, 2009, the licensee identified, for a postulated fire that requires a plant shutdown with 20 minute depressurization, the licensee's SSD procedures did not include steps to reestablish required Appendix R system alignments in the event that reactor vessel water level falls below the Reactor Vessel Water Level 1 setpoint. This condition has existed since the BFN Safe Shutdown analysis established the site's SSD methodology in January 1986. Upon discovery, the licensee entered the condition into the corrective action program (PER 177130) and implemented compensatory measures to inhibit the accident logic after SSI entry. The licensee also implemented fire watches as additional compensatory measures.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for this issue in accordance with the NRC Enforcement Policy, Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" and Inspection Manual Chapter 0305, Operating Reactor Assessment Program, Section 11.05.b. Specifically, this issue was identified and will be addressed during the licensee's transition to NFPA 805, it was entered into the licensee's corrective action program, immediate corrective action and compensatory measures were taken, it was not likely to have been previously identified by routine licensee efforts, it was not willful, and it was not associated with a finding of high safety significance (Red).

.4 (Closed) Licensee Event Report (LER) 05000259, 260, 296/2012-004-00, and 01: Fire Damage to Cables in Fire Areas Could Cause a Residual Heat Removal Service Water Pump to Spuriously Start

a. Inspection Scope

The inspectors reviewed LER 2012-004-00 and -01 that documented a deficiency in the fire protection program. The LER documented discovery of a condition where a postulated fire could result in the spurious start of a RHRSW pump. This condition was identified during the licensee's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed in-plant walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee's compensatory measures and corrective actions.

The LERs are closed.

b. Findings

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

Description. On May 11, 2012, the licensee submitted LER 2012-004-00, which documented discovery of a condition where a postulated fire could result in the spurious start of a RHRSW pump. This condition was identified during the licensee's transition to NFPA 805. During the review process, the licensee discovered that fire damage to the limit switches or relays for RHRSW pumps A1, B1, C1, D1 could cause these pumps to spuriously start upon an Emergency Equipment Cooling Water (EECW) automatic initiation signal. The spurious start of these pumps could result in overloading, and possible failure, of the credited emergency diesel generator. This condition could adversely impact the capability to supply power to credited Appendix R safe shutdown equipment. This condition is applicable to postulated fires in FZ 2-4 (Unit 2 Reactor Building, EL 593 south of column line Q and RHR Heat Exchanger Rooms, EL.565 and 593), FZ 3-3 (Unit 3 Reactor Building, EL 593, and RHR heat exchanger rooms, EL 565 and 593 near column R15-S and R21-S.), and FA 25-1 (Intake Pumping Station and Cable Tunnel to Turbine Building to Fire Door 440, and IPS EL 565 Deck and RHRSW Compartments B and D). While performing the extent of condition review, the licensee determined that RHRSW pumps A3, B3, C3, and D3 have similar circuitry. As a result, these pumps could spuriously start for postulated fires in 7 additional fire areas. The licensee submitted a revision to the LER on December 26, 2012 to reclassify the issue as not being a Safety System Functional Failure.

The licensee determined that the deficiencies existed because of human performance errors by engineers. This LER was applicable to Units 1, 2 and Unit 3. Upon discovery, the licensee entered this issue into their corrective action program as PER 521739, and implemented compensatory actions in the form of fire watches for the affected FAs. Subsequently, the licensee implemented a modification to reconfigure the wiring of the RHRSW pumps to eliminate the potential of EDG overloading due to RHRSW pump start.

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

NFPA 805, this issue is being dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy.

In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less-than-Red (i.e. ΔCDF less than 1E-4/year). The inspectors also performed walkdowns to verify key assumptions were applicable. Based on the ignition frequency of fire sources in the affected areas, inspectors determined that the significance of this non-compliance was less-than-Red. The inspectors also noted that the values in the licensee's quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red).

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

<u>Enforcement</u>. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G. 10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:
a) separation of cables and equipment by a fire barrier having a 3-hour rating;
b) separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area; or c) enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of the means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on June 1, 2012, and July 11, 2012, the licensee identified the failure to protect equipment that was required to mitigate fire events. The licensee determined that fire damage could cause mal-operation of RHRSW pumps, potentially leading to the overloading of the EDG credited for SSD. This condition has existed since initial plant startup for Units 1, 2 and 3. The licensee entered this issue into the corrective action program (PER 521739) and implemented compensatory actions in the form of fire watches for FZs 2-4, 3-3, and 25-1.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for this issue in accordance with the NRC Enforcement Policy, Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" and Inspection Manual Chapter 0305, Operating Reactor Assessment Program, Section 11.05.b. Specifically, this issue was identified and will be addressed during the licensee's transition to NFPA 805, it was entered into the licensee's corrective action program, immediate corrective action and compensatory measures were taken, it was not likely to have been previously identified by routine licensee efforts, it was not willful, and it was not associated with a finding of high safety significance (Red).

.5 <u>(Closed) Licensee Event Report (LER) 05000260, 296/2012-005-00: Unanalyzed Conditions Discovered during National Fire Protection Association 805 Transition Affecting Division II of the Residual Heat Removal (RHR) System</u>

a. <u>Inspection Scope</u>

The inspectors reviewed LER 2012-005-00 that documented a deficiency in the fire protection program. The LER documents that certain postulated Unit 2 and 3 fires could result in fire induced damage of control circuits associated with motor operated valves in the RHR system. The licensee identified the deficiency during the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee's compensatory measures and corrective actions.

This LER is closed.

b. <u>Findings</u>

<u>Introduction</u>. A licensee-identified non-compliance with 10 CFR 50, Appendix R, Section III.G.2 was identified for the licensee's failure to protect one of the redundant trains of equipment needed to achieve post-fire SSD from fire damage. Specifically, the license failed to provide adequate protection for the control circuits of 2 motor operated valves (MOVs) that were utilized in safe shutdown procedures to mitigate fire events.

<u>Description</u>. On December 17, 2012, the licensee submitted LER 2012-005-00, which described a previously unanalyzed condition concerning fire protection. The LER documents that certain postulated fires could result in fire induced circuit damage of MOV control circuits for low pressure coolant injection valves 2-FCV-074-0067 and 3-FCV-074-0067 in the RHR system. These normally closed valves are required to be opened during fire events. The licensee determined that fire damage to the control circuitry could inhibit the ability of the valve to be opened (Unit 3); result in the spurious

closure of the valve (Unit 2 and 3); and adversely affect the close limit switch and/or torque switch functions (Unit 2 and 3). Failure of the limit/torque switch could subject the MOVs to forces that exceed design limits which could prevent the valves from being opened electrically or manually. The licensee entered this issue into their corrective action program as PERs 626885 and 626886, and implemented compensatory actions in the form of fire watches for Units 2 and 3. The licensee discovered the deficiencies during the review of their Appendix R design bases as part of the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

Analysis. The licensee's failure to protect one train of cables and equipment necessary to achieve post-fire SSD from fire damage for fire areas designated in the Fire Protection Report (FPR) as meeting Appendix R, Section III.G.2, was a performance deficiency. The performance deficiency was more than minor because it adversely affected the Mitigating Systems cornerstone attribute of Protection Against External Events (i.e., fire) and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to provide adequate circuit protection for the control circuits of valves 2-FCV-074-0067 and 3-FCV-074-0067 that were utilized in safe shutdown procedures to mitigate fire events. Because this issue relates to fire protection, and this non-compliance was identified as a part of the site's transition to NFPA 805, this issue was dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy.

In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less-than-Red (i.e. ΔCDF less than 1E-4/year). The inspectors also performed walkdowns to verify key assumptions were applicable. Based on the ignition frequency of fire sources in the affected areas, combined with the probability of non-suppression for those fire scenarios, inspectors determined that the significance of this non-compliance is less-than-Red. Inspectors also noted that the values in the licensee's quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red).

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

Enforcement. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G. 10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided: a) separation of cables and equipment by a fire barrier having a 3-hour rating; b) separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area; or c) enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on December 17, 2012, the licensee identified the failure to protect the control circuitry of MOVs 2-FCV-074-0067 and 3-FCV-074-0067 that were required to mitigate fire events. The licensee determined that fire damage could cause maloperation of the redundant trains of equipment due to hot shorts, open circuits, or shorts to ground. This condition has existed since initial plant startup for Units 2 and 3. The licensee entered this finding into the corrective action program (PERs 626885 and 626886) and implemented compensatory actions in the form of fire watches for Units 2 and 3.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for this issue in accordance with the NRC Enforcement Policy, Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" and Inspection Manual Chapter 0305, Operating Reactor Assessment Program, Section 11.05.b. Specifically, this issue was identified and will be addressed during the licensee's transition to NFPA 805, it was entered into the licensee's corrective action program, immediate corrective action and compensatory measures were taken, it was not likely to have been previously identified by routine licensee efforts, it was not willful, and it was not associated with a finding of high safety significance (Red).

.6 <u>(Closed) Licensee Event Report (LER) 05000259, 260, 296/2012-007-00, -01, and -02: Cable Routing Errors Found in the Appendix R Separation Analysis</u>

a. <u>Inspection Scope</u>

The inspectors reviewed LER 2012-007-02 that documented a deficiency in the fire protection program. The LER documented a cable routing error that was identified during the licensee's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed in-plant walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee's compensatory measures and corrective actions.

The LERs are closed.

b. Findings

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

Description. On July 31, 2012, the licensee submitted LER 2012-007-00, which documented a cable routing error that was identified during the licensee's transition to NFPA 805. During the review process, the licensee discovered that a cable routing error could result in failure of the direct current (DC) control power to dedicated 4kV Shutdown Board 3EA for a postulated fire in FA 23 (4kV Shutdown Board Room 3EC and 3ED). The licensee determined cables that provide the normal DC control power to Shutdown Board 3EA, as well as cables that provide alternate DC control power were both routed through FA 23. Because of this condition, the current Appendix R separation analysis incorrectly credits Shutdown Board 3EA as being able to perform its function for a postulated fire in FA 23.

Subsequently, the licensee submitted a revision to the LER on September 7, 2012. This revision documented an additional cable routing error that was discovered. The licensee discovered that cables for cross-tie breakers between 4kV shutdown busses B and 3EB are routed through FZ 3-2 (Unit 3 Reactor Building, EL 519 through 565, from column line R21 to 10 ft west of column line R18). Due to this condition, a postulated fire in FZ 3-2 could cause spurious closure of breakers 1848 and 1828, which could result in paralleling EDG B and EDG 3B out of phase.

The licensee determined that the deficiencies existed because of the lack of an effective program for technical human performance tools during the performance of the Appendix R separation analysis. The licensee submitted a subsequent revision to the LER on December 26, 2012, to reclassify the issue as not being a Safety System Functional Failure. This LER was applicable to Units 1, 2 and Unit 3. The licensee entered this issue into their corrective action program as PER 561101 and PER 577583, and implemented compensatory actions in the form of fire watches for FA 23 and FZ 3-2.

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

In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less-than-Red (i.e. ΔCDF less than 1E-4/year). The inspectors also performed walkdowns to verify key assumptions were applicable. Based on the ignition frequency of fire sources in the affected areas, inspectors determined that the significance of this non-compliance was less-than-Red. The inspectors also noted that the values in the licensee's quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red). The inspectors determined that no cross cutting aspect was applicable to this performance deficiency because this finding was not indicative of current licensee performance.

<u>Enforcement</u>. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G. 10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:
a) separation of cables and equipment by a fire barrier having a 3-hour rating;
b) separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area; or c) enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on June 1, 2012, and July 11, 2012, the licensee identified the failure to protect equipment that was required to mitigate fire events. The licensee determined that fire damage could prevent operation or cause mal-operation of the multiple 4kV shutdown boards due to hot shorts, open circuits, or shorts to ground. This condition has existed since initial plant startup for Units 1, 2 and 3. The licensee entered this issue into the corrective action program (PER 561101 and PER 577583) and implemented compensatory actions in the form of fire watches for FA 23 and FZ 3-2. Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for this issue in accordance with the NRC Enforcement Policy, Section 9.1, “Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)” and Inspection Manual Chapter 0305, Operating Reactor Assessment Program, Section 11.05.b. Specifically, this issue was identified and will be addressed during the licensee’s transition to NFPA 805, it was entered into the licensee’s corrective action program, immediate corrective action and compensatory measures were taken, it was not likely to have been previously identified by routine licensee efforts, it was not willful, and it was not associated with a finding of high safety significance (Red).

.7 (Closed) Licensee Event Report (LER) 05000259, 260, 296/2012-009-00: 480 Volt Shutdown Board Breaker Actions in Safe Shutdown Instruction Procedures May Not Work as Written Due to Cable Fire Damage

a. Inspection Scope

The inspectors reviewed LER 2012-009-00 that documented a deficiency in the fire protection program. The LER documents that certain postulated Unit 1, 2 and 3 fires could result in fire induced damage of control circuits associated with feeder breakers to the 480 volt shutdown boards. The licensee identified the deficiency during the site’s transition to NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.” The inspectors reviewed information contained in the LER and associated corrective action program documents to determine if a violation of regulatory requirements occurred; and reviewed qualitative and quantitative risk analyses performed by the licensee to verify that the finding was not of high safety significance (Red). Additionally, the inspectors performed walkdowns to verify key assumptions were applicable. The inspectors also assessed the adequacy of the licensee’s compensatory measures and corrective actions.

This LER is closed.

b. Findings

Introduction. A licensee-identified non-compliance with 10 CFR 50, Appendix R, Section III.G.2, was identified for the licensee’s failure to protect one of the redundant trains of equipment needed to achieve post-fire SSD from fire damage. Specifically, the licensee failed to provide adequate protection for the control circuits associated with the feeder breakers for the 480 volt shutdown boards that were utilized in safe shutdown procedures to mitigate fire events.

Description. On December 28, 2012, the licensee submitted LER 2012-009-00, which described a previously unanalyzed condition concerning fire protection. The LER documents that certain postulated fires could result in fire induced circuit damage of control circuits associated with feeder breakers to the 480 volt shutdown boards. The shutdown boards provide electrical power to components required for safe shutdown. Fire damage to the control circuits could cause the normal and/or alternate feeder breakers to spuriously trip open, and also inhibit the capability to close the breakers remotely or locally.

The licensee entered this issue into their corrective action program as PER 635230, implemented compensatory actions in the form of fire watches, and developed a modification (DCN 70880, Add Fuses and Revise 43 Switch Logic to Assure Control Power During Appendix R Events) to re-design the associated control circuitry. At the time of inspection, the modification was completed on Unit 2; and scheduled to be implemented during the upcoming refueling outages for Units 1 and 3. The licensee discovered the deficiencies during the review of their Appendix R design bases as part of the site's transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

Analysis. The licensee's failure to protect one train of cables and equipment necessary to achieve post-fire SSD from fire damage for fire areas designated in the Fire Protection Report (FPR) as meeting Appendix R, Section III.G.2, was a performance deficiency. The performance deficiency was more than minor because it adversely affected the Mitigating Systems cornerstone attribute of protection against external events (i.e., fire) and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to provide adequate circuit protection for the control circuits associated with the feeder breakers for the 480 volt shutdown boards that were utilized in safe shutdown procedures to mitigate fire events. Because this issue relates to fire protection, and this non-compliance was identified as a part of the site's transition to NFPA 805, this issue was dispositioned in accordance with Section 9.1, "Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)" of the NRC Enforcement Policy.

In order to verify that this non-compliance was not associated with a finding of high safety significance (Red), inspectors reviewed qualitative and quantitative risk analyses performed by the licensee. These risk evaluations took ignition source and target information from the ongoing BFN fire PRA to demonstrate that the significance of the non-compliances were less-than-Red (i.e. ΔCDF less than 1E-4/year). The inspectors also performed walkdowns to verify key assumptions were applicable. Based on the ignition frequency of fire sources in the affected areas, inspectors determined that the significance of this non-compliance is less-than-Red. Inspectors also noted that the values in the licensee's quantitative analysis were conservative, in that they used screening values instead of more detailed values. This provided additional confidence that this non-compliance was not associated with a finding of high safety significance (Red).

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

Enforcement. 10 CFR Part 50.48(b)(1) requires that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G. 10 CFR 50, Appendix R, Section III.G.2, states, in part, that where cables or equipment, that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided: a) separation of cables and equipment by a fire barrier having a 3-hour rating; b) separation of cables and equipment by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards and with fire detectors and an automatic fire suppression system in the fire area; or c) enclosure of cables and equipment in a fire barrier having a 1-hour rating and with fire detectors and an automatic fire suppression system in the fire area.

Contrary to the above, the licensee failed to use one of means described in Appendix R, Section III.G.2.a, b, or c to ensure that one of the redundant trains of equipment necessary to achieve and maintain hot shutdown conditions was protected from fire damage. Specifically, on December 28, 2012, the licensee identified the failure to protect the control circuits associated with the feeder breakers for the 480 volt shutdown boards that were required to mitigate fire events. The licensee determined that fire damage could result in the inability to energize required safe shutdown equipment due to open circuits caused by fire damage. This condition has existed since initial plant startup for Units 1, 2 and 3. The licensee entered this finding into the corrective action program (PER 635230), implemented compensatory actions in the form of fire watches, and developed a modification.

Because the licensee committed to adopt NFPA 805 and change their fire protection licensing bases to comply with 10 CFR 50.48(c), the NRC is exercising enforcement and reactor oversight process (ROP) discretion for this issue in accordance with the NRC Enforcement Policy, Section 9.1, “Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)” and Inspection Manual Chapter 0305, Operating Reactor Assessment Program, Section 11.05.b. Specifically, this issue was identified and will be addressed during the licensee’s transition to NFPA 805, it was entered into the licensee’s corrective action program, immediate corrective action and compensatory measures were taken, it was not likely to have been previously identified by routine licensee efforts, it was not willful, and it was not associated with a finding of high safety significance (Red).

.8 (Closed) Licensee Event Report (LER) 05000259, 260, 296/2013-003-00, ‘A’ Emergency Diesel Generator

a. Inspection Scope

The licensee event report was reviewed. A special inspection was conducted concerning the events of this LER and was documented in inspection report 05000259, 260, 296/2013405 issued July 17, 2013. This LER contained security sensitive information. Additional information is withheld from public disclosure in accordance with 10 CFR 2.390(d)(1).

This LER is closed.

b. Findings

No findings were identified.

.9 (Closed) Licensee Event Report (LER) 05000260/2013-001-00, RCIC Turbine Exhaust Stop-Check Valve Inoperable

a. Inspection Scope

On March 2, 2013, Unit 2 Reactor Core Isolation Cooling (RCIC) system was declared inoperable when radiographic testing discovered that 2-HCV-071-0014, RCIC Turbine Exhaust Stop-Check Valve, was stuck in the open position. The stop-check valve was one of two Primary Containment Isolation Valves (PCIVs) and provided isolation between the RCIC exhaust system and Primary Containment. The valve was declared inoperable per Technical Specification (T.S.) 3.6.1.3., Primary Containment Isolation Valves (PCIVs). As a result of this action the RCIC system was declared inoperable and the licensee entered Condition A of T.S. 3.5.3, RCIC System. Required Action A.2 required the RCIC system to be returned to operable status within 14 days. Since 2-HCV-071-0014 was unisolable a Unit 2 shutdown was required to repair the stuck stop-check valve. Following the unit shutdown, the licensee performed repairs on the valve. The licensee determined that the root cause for the stuck valve condition was improper classification as a Run-to-Failure hand valve. The licensee entered the issue into the corrective action program (CAP) as PER 691891and completed actions to change the reliability classification and require periodic inspection and maintenance of the valve internals. The inspectors reviewed the LER and the associated PER. The inspectors verified that the reliability classification for these valves has been changed to Critical and that periodic inspections have been incorporated into the outage schedules.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On October 4, 2013, the resident inspectors presented the quarterly inspection results to Mr. Keith Polson, 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.

On November 12, 2013, the resident inspectors provided a re-exit of the quarterly inspection results to Mr. Jamie Paul, Nuclear Site Licensing Manager who acknowledged the changes.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
E. Bates, Licensing
J. Boyer, Assistant Engineering Director
W. Byrne, Security Manager
D. Campbell, Assistant Ops Superintendent
P. Campbell, System Engineer
P. Donahue, Assistant Engineering Director
J. Emens, Nuclear Site Licensing Manager
D. Ford, System Engineer
A. Gallegos, System Engineer
D. Green, Licensing
R. Guthrie, System Engineer
L. Hughes, Manager Operations
D. Jackson, System Engineer
J. Lacasse, System Engineer
T. Mingus, Engineering
S. Norris, Components Mgr.
M. Oliver, Licensing
J. Paul, Nuclear Site Licensing Manager
K. Polson, Site Vice President
M. Roy, System Engineer
S. Samaras, Civil Design Engineer
V. Schiavone, BWRVIP Program Owner
L. Soto, Engineering
C. Vaughn, Ops Training Manager
M. Webb, Site Licensing
S. Wentzel, System Engineer
A. Yarborough, System Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Opened and Closed

05000259,260,296/2013004-01	NCV	Failure to enter Technical Specification for Residual Heat Removal Service Water Maintenance (Section 1R13)
05000259,260,296/2013004-02	FIN	Failure to Clean the Safety Related Pump Pit Once per Two Cycles (Section 1R15)
05000296/2013004-04	FIN	Failure to Properly Screen and Classify Corrective Action Program, Problem Evaluation Reports (Section 4OA3.3)

Closed

05000259/2013-002-00	LER	Manual Reactor Shutdown Due to Decreasing Vacuum (Section 4OA3.1)
05000296/2013-003-00	LER	Automatic Reactor Shutdown Due to an Actuation of the Reactor Protection System from a Turbine Trip (Section 4OA3.2)
05000259,260,296/2009-005-00	LER	Reactor Vessel Water Level 1 Initiation Logic Including the Common Accident Logic Not Evaluated for Appendix R Fire Event (Section 4OA3.3)
05000259,260,296/2009-005-01	LER	Reactor Vessel Water Level 1 Initiation Logic Including the Common Accident Logic Not Evaluated for Appendix R Fire Event (Section 4OA3.3)
05000259,260,296/2009-005-02	LER	Reactor Vessel Water Level 1 Initiation Logic Including the Common Accident Logic Not Evaluated for Appendix R Fire Event (Section 4OA3.3)
05000259,260,296/2012-004-00	LER	Fire Damage to Cables in Fire Areas Could Cause a Residual Heat Removal Service Water pump to Spuriously Start (Section 4OA3.4)

05000259,260,296/2012-004-01	LER	Fire Damage to Cables in Fire Areas Could Cause a Residual Heat Removal Service Water Pump to Spuriously Start (Section 4OA3.4)
05000260,296/2012-005-00	LER	Unanalyzed Conditions Discovered during National Fire Protection Association 805 Transition Affecting Division II of the Residual Heat Removal (RHR) System (Section 4OA3.5)
05000259,260,296/2012-007-00	LER	Cable Routing Errors Found in the Appendix R Separation Analysis (Section 4OA3.6)
05000259,260,296/2012-007-01	LER	Cable Routing Errors Found in the Appendix R Separation Analysis (Section 4OA3.6)
05000259,260,296/2012-007-02	LER	Cable Routing Errors Found in the Appendix R Separation Analysis (Section 4OA3.6)
05000259,260,296/2012-009-00	LER	480 Volt Shutdown Board Breaker Actions in Safe Shutdown Instruction Procedures May Not Work as Written Due to Cable Fire Damage (Section 4OA3.7)
05000259/,260,296/2013-003-00	LER	‘A’ Emergency Diesel Generator (Section 4OA3.8)
05000260/2013-001	LER	RCIC Turbine Exhaust Stop-Check Valve Inoperable (Section 4OA3.9)
05000259,260,296/2013-003-02	URI	LER 05000259/2013-002-00 (Section 4OA3.2) and LER 05000296/2013-003-00 (Section 4OA3.2)
05000260/2013-003-03	URI	Residual Heat Removal (RHR) Heat Exchanger (HX) Excessive Fouling (Section 1R15)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment
Offsite Dose Calculation Manual (ODCM) Chapter 7
Updated Final Safety Analysis Report 5.3 Secondary Containment
Technical Specification 3.6.4.3 Standby Gas Treatment System
Technical Specification 5.5.7 Ventilation Filter Testing Program (VFTP)
Technical Specification Basis 3.6.4.3
Drawing 0-47E771-2 480V Diesel Auxiliary Power
Surveillance 0-SR-3.6.4.3.2 (C VFTP)
Surveillance 0-SR-3.6.4.3.2 (C Heater)
Surveillance 0-SR-3.6.4.3.2 (C) Iodine Removal Efficiency
G-55 Technical Requirements for Protective Coatings for TVA Nuclear Plants; Rev 19
MAI-5.3 Protective Coatings for Service Level I, II, and III and Corrosive Environments; Rev 49
1-OI-75/ATT-1 Core Spray System Valve Lineup Checklist, Rev. 21
1-OI-75/ATT-2 Core Spray System Panel Lineup Checklist, Rev. 21
1-OI-75/ATT-3 Core Spray System Electrical Lineup Checklist, Rev. 21
1-47E814-1 Core Spray System Flow Diagram, Rev. 23
Design Criteria Document, Core Spray Cooling System, Rev. 12

Section 1R05: Fire Protection
Fire Protection Report Volume 1, Rev 15
Fire Protection Report Volume 1, Section 2, Fire Hazard Analysis, Part 6.0, Detailed Fire Prevention Protection Evaluations, Fire Areas 1-3 and 1-4, Rev. 15
Fire Protection Report Volume 1, Section 3, Safe Shutdown Analysis, Part 8.0, Availability of Minimum SSDS for Individual Fire Areas/Zones, Fire Areas 1-3 and 1-4, Rev. 15
Fire Protection Report Volume 2, Appendix O, Section IV, Pre-Fire Plans for Browns Ferry Nuclear Plant – Reactor Building Unit 1 EL 593', Pre-Plan No. RX1-593, Rev. 51
Fire Protection Report Volume 2, Rev 51
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WO 114087239, Perform 3-SR-3.1.4.1 Scram Insertion Times

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BFN-0-12-146, PRA Evaluation Response, Cable Failure Could Cause Loss of the 480V Shutdown Boards, dated 11/14/12

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LER 50-260/2012-005-00, Unanalyzed Conditions Discovered during NFPA 805 Transition Affecting Division II of the Residual Heat Removal System, dated 12/17/12
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PER 511932, 2 gpm (approximate) leak from unidentified location on 2-FCV-3-77

PER 514913, Need PER to track ODMI for 2-FCV-3-73 long cycle valve
PER 521739, Appendix R DG Loading Issue – RHRSW Pump Spurious Start
PER 561101, Appendix R cable routing error on cable 3B181
PER 577583, Appendix R cable PP698-IB is routed in Fire Zone 3-2
PER 600143, Piping leak downstream of long cycle valve. (previously under SR 597256)
PER 600927, Unit 3 long-cycle piping leak risk decision making required QA intervention for ODMI request
PER 604937, Recent thru wall leak on U3 long cycle line should be reviewed for all units
PER 626885, Fire Damage Could inhibit the Ability to Open the Credited LPCI Injection Valve, 3-FCV-74-0067
PER 626886, Fire Damage Could Cause a Spurious Closure of 2-FCV-074-0067 or 3-FCV-074-0067
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PER 703283, REC 687732 on U3 Vacuum Loss Scram OE Preventable
PER 703283, REC 687732 on U3 Vacuum Loss Scram OE Preventable
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SR 597256, Thru wall leak downstream of Long cycle valves
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SR 604304, Replace U3 Long Cycle Feedwater piping during the U316 RFO
SR 727450, WO to Install External Thermocouples
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WO 113262348, See Revision 1. Install Furmanite box on 8 inch line downstream of -2-FCV-003-0073
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WO 113448971, Perform thermography on Long Cycle FCVs to determine if valves leak by
WO 113809831, Install a furmanite box Thru wall leak downstream of Long cycle valves
WO 113869582, Replace U3 Long Cycle Feedwater piping during the U316 RFO
WO 114130202, BFN-2-BDBB-231-0002A, Implement DCN 70880 Stage 3, dated 2/18/13
WO 114130203, BFN-2-BDBB-231-0002B, Implement DCN 70880 Stage 4, dated 2/20/13
WO 114391756, Perform permanent repair to 8" Misc Drain header in MS Room
WO 114392647, Perform UT readings of 8" Feedwater Long Cycle Line in MS Room.
WO 114512929, Repair / Replace Rod Hanger BFN-1-MISC-006
WO 114516358, Steam leak identified on the 4" Misc Dr Header weld-o-let to the 24" 1C2 Heater High Level Dump line
WO 114522755, Replace BFN-1-DRV-006-0557
WO 114522757, Replace 1-DRV-006-0612, 1C1 Tube Drain Valve
WO 114522759, Replace 1C3 Tube Side Relief Valve BFN-1-RFV-006-0652
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LIST OF ACRONYMS

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