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

April 27, 2012

Mr. J.W. Shea
Manager, Corporate Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 4B-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2012002, 05000260/2012002, 05000296/2012002 and 07200052/2012001

Dear Mr. Shea:

On March 31, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. The enclosed inspection report documents the inspection results which were discussed on April 6, 2012, with Mr. Lang Hughes and other members of your staff.

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

Three NRC identified findings of very low safety significance (Green) were identified during this inspection. These findings were determined to involve violations of NRC requirements. Additionally, the NRC has determined that a traditional enforcement Severity Level IV violation occurred. Furthermore, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. The NRC is treating the violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to: (1) the Regional Administrator, Region II; (2) the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and (3) the NRC Resident Inspector at the Browns Ferry Nuclear Plant.

In addition, if you disagree with any cross-cutting aspect assignment in the report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Browns Ferry Nuclear Plant.

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

Sincerely,

/RA/

Eugene F. Guthrie, Chief
Special Project, Browns Ferry
Division of Reactor Projects

Docket Nos.: 50-259, 50-260, 50-296, 72-052
License Nos.: DPR-33, DPR-52, DPR-68

Enclosure: NRC Integrated Inspection Report 05000259/2012002, 05000260/2012002, 05000296/2012002, and 07200052/2012001

cc w/encl. (See page 3)

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Letter to Joseph W. Shea from Richard P. Croteau dated April 27, 2012

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2012002, 05000260/2012002, 05000296/2012002 and 07200052/2012001

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296, 072-052
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2012002, 05000260/2012002, 05000296/2012002, and 07200052/2012001
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	January 1, 2012, through March 31, 2012
Inspectors:	C. Stancil, Resident Inspector P. Niebaum, Resident Inspector L. Pressley, Resident Inspector K. Korth, Senior Training Instructor (1R04, 1R12, 1R13) R. Carrion, Senior Reactor Inspector (4OA5.2) C. Fletcher, Senior Reactor Inspector (4OA5.2)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Special Branch Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259/2012002, 05000260/2012002, 05000296/2012002, 07200052/2012001; 01/01/2012 –03/31/2012; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Fire Protection, Event Follow-up, Other Activities

The report covered a three month period of inspection by resident and regional inspectors. Three non-cited violations (NCVs) were identified. The significance of most findings is identified by their color (Green, White, Yellow, and Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); 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.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The NRC identified a non-cited violation of Technical Specification 5.4.1.d, Fire Protection Program, for the licensee's failure to adequately implement Limiting Conditions For Operation in accordance with Fire Protection Report Volume 1, Fire Protection Plan. Specifically, the licensee failed to adequately implement impaired fire barrier and detector controls which resulted in the failure to establish a continuous fire watch for an impaired fire barrier having smoke detection identified as unavailable to protect either side of the inoperable barrier. The licensee subsequently returned the impaired fire door and smoke detection to service. The licensee entered this event into their corrective action program as PERs 529543 and 527311.

  The finding was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Protection Against External Events, and adversely affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, inadequate implementation of the licensee's FPIP and LCO processes resulted in the licensee missing a LCO entry condition and not establishing a continuous fire watch for an impaired fire door. The significance of this finding was evaluated in accordance with the IMC 0609 Appendix F, Attachment 01, Part 1, Fire Protection SDP Phase 1 Worksheet. The finding was determined to be of very low safety significance (Green) because the condition represented a low degradation of fire prevention and administrative controls. Specifically, a smoke detection system on one side of the impaired fire door was discovered functional. The cause of this finding was directly related to the cross cutting aspect of Procedural Compliance in the Work Practices component of the Human Performance area, because licensee expectations were ineffectively communicated and fire protection procedures inadequately implemented to maintain a site understanding of fire barrier and detector configuration [H.4(b)]. (Section 1RO5)

Cornerstone: Initiating Events

- Green. The NRC identified a non-cited violation of Technical Specification 5.4.1.d, Fire Protection Program implementation associated with the licensee's failure to report a fire in the Unit 1 Turbine Building to the main control room (MCR). Specifically, the failure to report a plant fire resulted in a failure of the MCR operators to implement Emergency Plan Implementing Procedure EPIP-17, Fire Emergency Response. Following the event, plant staff performed additional inspections of plant areas and either removed electrical extension cords or ensured each cord had a required GFCI and was not overloaded. Expectations for plant employees discovering and responding to fires were reinforced by plant management. The licensee entered this event into their corrective action program as PER 527090.

  The performance deficiency was determined to be more than minor because if left uncorrected, the failure to notify the MCR of plant fire events would have the potential to lead to a more significant safety concern. Specifically, emergency response procedures for plant fires would not be entered and implemented and the Fire Brigade response would be delayed. The significance of this finding was evaluated in accordance with the IMC 0609, Appendix F, Attachment 1, Part 1, Fire Protection SDP Phase 1 Worksheet. The inspectors concluded that the significance of this finding was Green due to a low degradation rating for this fire event because it was a small electrical fire with no combustible material within the vicinity of the fire. The cause of this finding was directly related to the cross cutting aspect of Procedural Compliance in the Work Practices component of the Human Performance area, because the licensee failed to recognize the requirement to immediately report a fire and enter the appropriate fire emergency response procedures [H.4(b)]. (Section 4OA3.4)

Cornerstone: Barrier Integrity

- Green: The NRC identified a Green non-cited violation of 10CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, for the licensee's failure to ensure that the ECCS was satisfactorily designed such that the maximum fuel element cladding temperature specified in 10 CFR 50.46(b)(1) would not be exceeded. On May 29, 2011, operating limitations were implemented to account for the error in calculations. This violation has been entered into the licensee's CAP as PER 372764.

  This performance deficiency was considered greater than minor because it was associated with the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents. The inspectors determined the finding to not be greater than green based on the remaining barriers to fission product release were unaffected. The cause of this finding was directly related to the cross-cutting aspect of Issue Identification in the Corrective Action Program component of the Problem Identification and Resolution area because the licensee failed to completely, accurately, and in a timely manner identify the errors with the ECCS evaluation model [P.1.(a)]. (4OA5.3)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full Rated Thermal Power (RTP) for the report period.

Unit 2 operated at essentially full RTP for most of the report period except for a planned downpower on March 8 to 92 percent RTP power to remove the C1 and C2 high pressure feedwater heaters from service to facilitate downstream piping repairs. The unit returned to full RTP the same day.

Unit 3 operated at essentially full RTP for most of the report period except for two unplanned downpowers. On January 21, the unit performed an unplanned downpower to 80 percent RTP due to false temperature indications on the main generator's stator cooling water system. The unit returned to full RTP on January 22. On February 24, the unit performed an unplanned downpower to 43 percent RTP because the 3A condenser circulating water (CCW) pump tripped while the 3C CCW pump was out of service for planned maintenance. Unit 3 returned to full RTP on February 29.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Tornado Warning

a. Inspection Scope

On March 2, a Tornado Warning was issued for adjacent counties. The inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions and observed the licensee's implementation of abnormal operating instruction 0-AOI-100-7, Severe Weather. The inspectors also reviewed and discussed the implementation of 0-AOI-100-7 with the responsible Unit Supervisors (US) and Shift Manager. Furthermore, the inspectors witnessed the licensee's execution of evacuation orders of vulnerable areas and buildings outside the power block, and the termination of work and evacuation of the turbine and refueling floors. The inspectors also toured the plant grounds for loose debris, which could become missiles during a tornado, and ascertained operator staffing and if they could access controls and indications for those systems required for safe control of the plant. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

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

- Unit 1/2 ‘A’ Emergency Diesel Generator (EDG)
- Unit 1 Residual Heat Removal (RHR) System, Division II
- Unit 3 Qualified Alternate Offsite Power Sources (161 kV) with Athens 161 kV Line Out of Service

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 1/2 C EDG, using the applicable P&ID flow diagrams, 0-47E861-3, 0-47E861-7, and 0-47E840-2, along with the relevant operating instructions, 0-OI-82, to verify equipment availability and operability. The inspectors reviewed relevant portions of the UFSAR and TS. This detailed walkdown also verified electrical power alignment, the condition of applicable system instrumentation and controls, component labeling, pipe hangers and support installation, and associated support systems status. Furthermore, the inspectors examined applicable System Health Reports, open Work Orders, and any previous Problem Evaluation Reports (PERs) that could affect system alignment and operability. This activity constituted one inspection sample.

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 the six 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. 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 inspection samples.

- Fire Area (FA) 5: Unit 1 Reactor Building, Electrical Board Room 1A and 250V Battery Room, EL 621'
- FA 17: Unit 1 Control Building EL 593' Battery and Battery Board Rooms
- Fire Zone (FZ) 2-3: Unit 2 Reactor Building, EL 593' and the RHR HX Room
- FZ 2-4: Unit 2 Reactor Building, EL 593' and the RHR HX Room
- FZ 2-5: Unit 2 Reactor Building, EL 621' and EL 639 North of Column Line R
- FA 25-1: Cable Tunnel to Intake Pumping Station and Fire Door 440

b. Findings

One finding was identified.

Introduction: The NRC identified a Green, non-cited violation of Technical Specification 5.4.1.d, Fire Protection Program, for the licensee's failure to adequately implement Limiting Conditions For Operation in accordance with Fire Protection Report Volume 1, Fire Protection Plan. Specifically, the licensee failed to adequately implement impaired fire barrier and detector controls which resulted in the failure to establish a continuous fire watch for an impaired fire barrier having smoke detection identified as unavailable to protect either side of the inoperable barrier.

Description: During a walk down of the intake pumping station cable tunnel on March 22, 2012, inspectors observed that Fire Door (FD)-440 was blocked with a scaffold board and temporary sump pump discharge hose protruding through the door opening. This blockage would have prevented the fusible link closure assembly from automatically closing the door during a fire event. FD-440 was a recent fire barrier modification implemented as a result of Browns Ferry's transition to NFPA 805 fire protection program. This fire door separated the turbine building fire area from the intake pumping station fire areas. The inspectors determined that the licensee initiated Fire Protection

Impairment Permits (FPIPs) 12-3398 and 12-3394 for FD-440 and the local linear beam smoke detectors respectively, as a result of ongoing replacement of condenser circulating water pump power supply cables on Feb 27, 2012. In addition, the inspectors determined that the licensee implemented FPIP 11-2888 for the system of smoke detectors (separate from the linear beam detectors) on both sides of FD-440, due to a licensee determination that associated fire protection control panel 297 was OOS on March 1, 2011. The inspectors concluded that, based on the fire protection equipment status, the cable tunnel to the intake pumping station was without an adequate fire barrier or detection and in accordance with Fire Protection Report Volume 1, Section 9.3, Fire Protection Systems Limiting Condition for Operating, the licensee should have established a continuous fire watch in the cable tunnel beginning February 27, 2012. After this issue was brought to the licensee's attention by the inspectors, and the determination that no work requiring the impairments was in progress, FPIP 12-3394 was restored to operation on March 23, 2012, and FPIP 12-3398 was restored to operation on March 22, 2012.

Upon follow-up functional testing, the licensee determined that fire panel 297 could respond to at least one system of tunnel smoke detectors and that the panel had been improperly designated OOS for over a year as a result of operators misunderstanding the panel functions. Also, even though the 297 panel was declared OOS, the limiting condition for operating (LCO) was never administratively entered on the FPIP and in the LCO log in accordance with Fire Protection Report, Section 9.3.11.G, which contributed to not establishing a continuous fire watch. In addition, operators exceeded the monthly surveillance grace period for fire panel 297, 1-SI-4.11.A.3, Monthly Functional Test of Non-Supervised Alarm Circuits, because operators had erroneously determined the panel as OOS. FPIP 11-2888 was restored to operation on March 28, 2012.

Furthermore, inspectors observed that site sensitivity to fire protection equipment availability was not consistent with minimizing OOS time as required by Fire Protection Report Volume 1, Fire Protection Plan, Section 7.4, Control of Fire Protection Impairments. All three FPIPs above were past their restoration dates, in one case, by almost a year. The inspectors also observed that the licensee procedure FP-0-000-INS019, Fire Protection Weekly Inspection, which required inspection of the impaired equipment to ensure accurate status, was ineffective in maximizing availability.

The licensee immediately restored the fire barrier and an associated detection system, and added senior licensed operator review of all fire impairments. In addition, the licensee plans to evaluate their FPIP and fire protection LCO programs to identify potential program improvements. The licensee entered this event into their corrective action program as PERs 529543 and 527311**.**

Analysis: The licensee's failure to adequately implement impaired fire barrier and detector controls in accordance with the fire protection program was a performance deficiency. The inspectors determined this finding to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Protection Against External Events, and adversely affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, inadequate implementation of the licensee's FPIP and LCO processes resulted in the licensee missing a LCO entry condition and not establishing a continuous fire watch for an impaired fire door.

The significance of this finding was evaluated in accordance with the IMC 0609 Attachment 4, Phase 1- Initial Screening and Characterization of Findings, which required further evaluation in accordance with Appendix F, Attachment 01, Part 1, Fire Protection SDP Phase 1 Worksheet. The finding was determined to be of very low safety significance (Green) because the condition represented a low degradation of fire prevention and administrative controls. Specifically, a smoke detection system on one side of the impaired fire door was discovered functional.

The cause of this finding was directly related to the cross cutting aspect of Procedural Compliance in the Work Practices component of the Human Performance area, because licensee expectations were ineffectively communicated and fire protection procedures inadequately implemented to maintain an adequate site understanding of fire barrier and detector configuration [H.4(b)].

Enforcement: Technical Specification 5.4.1.d required that written procedures shall be established, implemented, and maintained for Fire Protection Program implementation. Fire Protection Report Volume 1, Fire Protection Plan, Section 9.3.11.G, LCO for Fire Rated Assemblies, required that with FD-440 inoperable, a continuous fire watch on one side of the fire door was required to be established within one hour if no fire detection was available to protect either side of the inoperable fire door. Contrary to this requirement, the licensee failed to establish a continuous fire watch on one side of the door when no fire detection was available to protect either side of the inoperable fire door. Specifically, from February 27, to March 23, 2012, the licensee failed to enter an LCO entry condition which resulted in the failure to establish a continuous fire watch for impaired fire door, FD-440. The licensee immediately restored the fire barrier and an associated detection system. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PERs 529543 and 527311, this violation is being treated as an NCV consistent with the NRC Enforcement Policy. This NCV is identified as NCV 05000259, 260, 296/2012002-01, Failure to Adequately Implement Impaired Fire Barrier and Detector Controls.

.2 Annual Fire Brigade Drill

a. Inspection Scope

On February 8, 2012, the inspectors witnessed an unannounced fire drill in the Unit 1 Control Bay Elevation 593' at the Unit 1 Computer Room. The inspectors assessed fire alarm effectiveness; response time for notifying and assembling the fire brigade; the selection, placement, and use of firefighting equipment; use of personnel fire protective clothing and equipment (e.g., turnout gear, self-contained breathing apparatus); communications; incident command and control; teamwork; and fire fighting strategies. The inspectors also attended the post-drill critique to assess the licensee's ability to review fire brigade performance and identify areas for improvement. Following the critique, the inspectors compared their findings with the licensee's observations and to the requirements specified in the licensee's Fire Protection report. This activity constituted one inspection sample.

b. Findings

No findings were identified

1R06 Internal Flood Protection Measures

.1 Review of Areas Susceptible to Internal Flooding

a. Inspection Scope

The inspectors performed walkdowns of the internal flood protection features of three risk-significant areas in the Units 1, 2 and 3 Reactor Buildings (519' elevation) with susceptible systems and equipment, which included; Residual Heat Removal (RHR) and Core Spray (CS) pump rooms, High Pressure Coolant Injection (HPCI) pump rooms and Under-Torus areas. The inspectors reviewed selected licensee documents including: the UFSAR and design criteria; technical drawings; and procedures for mitigating and responding to flooding events, maintenance, testing, and annunciation response to verify that licensee actions were consistent with the plant's licensing and design basis.

The inspectors specifically examined plant design features and measures intended to protect the plant from an internal flooding event in any Reactor Building, such as Reactor Building bulkhead watertight doors, curbing, wall penetrations, and flood level and floor drain instrumentation. The inspectors also 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. Furthermore, the inspectors reviewed the PERs initiated for the previous 12 months with respect to flood-related items and to verify that problems were being identified and entered into the corrective action program. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On February 6, 2012, the inspectors observed a licensed operator requalification simulator training session for an operating crew according to Unit 2 Simulator Exercise Guide OPL177.041, H2 Supply Alarm, HPCI Pressure Switch Failure, Condenser Tube Leak, Fuel Failure, Main Steam Line leak, Unisolable RCIC Steam Line Break, HPCI

Failure, 2 Area Rad Levels Above Max Safe. Additionally, on February 19, 2012, the inspectors observed licensed operator requalification classroom and simulator sessions for an operating crew validating recently changed Safe Shutdown Instruction (SSI) 0-SSI-26, Turbine Bldg, Turbine Bldg Side of Cable Tunnel to Door 440, and Radwaste Building.

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 Abnormal Operating Instructions (AOIs), and Emergency Operating Instructions (EOIs)
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedures (EPIP)
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the Unit Supervisor and Shift Manager

The inspectors attended the post-examination critique to assess the effectiveness of the licensee evaluators, and to verify 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 constituted one inspection sample.

b. <u>Findings</u>

No findings were identified.

.2 <u>Control Room Observations</u>

a. <u>Inspection Scope</u>

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 inspection sample.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

.1 Routine

a. Inspection Scope

The inspectors reviewed the two 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-3.4, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; and NPG-SPP-03.1, Corrective Action Program. The inspectors also reviewed, as applicable, work orders, surveillance records, PERs, system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met. This activity constituted two inspection samples.

- Unit 1, 2 and 3 High Pressure Coolant Injection (HPCI) System Unavailability due to Steam Admission Valve leakage
- Unit 1 HPCI System Exceeded Unreliability Performance Criteria

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 five 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 (RMAs) were conducted as required by 10 CFR 50.65(a)(4) and applicable plant procedures such as NPG-SPP-7.0, Work Management; NPG-SPP-7.1, On-Line Work Management; 0-TI-367, BFN Equipment to Plant Risk Matrix; NPG-SPP-7.3, Work Activity Risk Management Process; and NPG-SPP-7.2, Outage Management. Furthermore, as applicable, the inspectors verified the actual in-plant configurations to ensure accuracy of the licensee's risk assessments and adequacy of RMA implementation. This activity constituted five inspection samples.

- Unit 1/2 C EDG and common B Control Bay Chiller out of service (OOS)
- 250 VDC Shutdown Board A Battery, B Control Bay Chiller, B Control Air Compressor, 3A Control Bay Chiller, 3B RCW Pump, 3B1 Shutdown Board Room Chiller OOS
- Unit 3 HPCI, Unit 1 Electric Board Room 1B AHU, Environmental factors for switchyard work, Red grid reliability rating, Unit 3 turbine trip multiplier for Stator Cooling Water Pump 3A OOS
- Unit 1 HPCI OOS, B Control Air Compressor, 1A Raw Cooling Water Pump, work in the switchyard, Trinity 161kV line switching
- South EECW Header, Main Bank 4 Battery and Charger OOS, Switchyard high risk for 500 kV (Pre-Outage Work) and 161 kV (PCB 928)

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the six 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, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors examined the implementation of compensatory measures to verify that they achieved the intended purpose and that the measures were adequately controlled. The inspectors also reviewed PERs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. This activity constituted six inspection samples.

- C EDG Overload Condition (PER 493979)
- Unit 2 RCIC 2-FCV-71-2, Inboard Isolation Valve electric backseat evaluation (PER 447881)
- C EDG Hole Drilled in Day Tank During Performance of DCN 69454 (PER 486972)
- 3C EDG Shorted Rotor Pole (PER 480886)
- Less than Rigorous Implementation of Technical Specification 5.5.2 Primary Coolant Sources Outside Containment (PER 468785)
- Unit 3 HPCI Stop Valve did not open after simulated overspeed trip test (PER 521782)

b. Findings

No findings were identified.

1R18 Plant Modifications

a. Inspection Scope

The inspectors reviewed the modifications listed below to verify regulatory requirements were met, along with procedures, as applicable, such as NPG-SPP-9.3, Plant Modifications and Engineering Change Control; NPG-SPP-9.5, Temporary Alterations; and NPG-SPP-6.9.3, Post-Modification Testing. The inspectors also reviewed the associated 10 CFR 50.59 screenings and evaluations and compared each against the UFSAR and TS to verify that the modifications did not affect operability or availability of the affected systems. Furthermore, the inspectors walked down each modification to ensure that it was installed in accordance with the modification documents and reviewed post-installation and removal testing to verify that the actual impact on permanent systems was adequately verified by the tests. This activity constituted two inspection samples.

- TACF-3-10-010-210, Revision 1, Replacement of One OOS Standby Emergency Diesel Generator (DG) with Two Temporary Standby DGs and Associated Circuits to Tie into the Bus-Tie Board
- DCN 69454, DG Turbocharger Lube Oil System Modification performed under Work Order 09-710573-000

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors witnessed and reviewed the six post-maintenance tests (PMT) listed below to verify that procedures and test activities confirmed SSC operability and functional capability following the described maintenance. The inspectors reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also 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 procedural requirements, including NPG-SPP-06.3, Pre-/Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors verified that problems associated with PMTs were identified and entered into the CAP. This activity constituted six inspection samples.

- Unit 1/2 C EDG per PMTI-69454-STG003, Post Modification Test Instruction for DCN 69454
- Unit 1 1C RHR heat exchanger to disassemble, clean and perform eddy current testing per WO 112806263 and 1-SR-3.5.1.6(RHR I), Quarterly RHR System Rated Flow Test Loop 1
- Unit 1 HPCI 1-FCV-073-0016 HPCI steam supply valve repairs per WO's 112347282, 112692321 and 113272032 and 1-SR-3.6.1.3.5(HPCI), HPCI System Motor Operated Valve
- Unit 2 Replacement of 2B Reactor Feed Pump Controller 2-SIC-046-0009 per WO 113144298
- Unit 3 3C EDG AC Pole Drop and Impedance Testing per WOs 112486450 and 112486454 and 3-SR-3.8.1.1(3C), Diesel Generator 3C Monthly Operability Test
- Unit 3 HPCI system outage per WO 111847024 and 3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure

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 five surveillance tests of risk-significant and/or safety-related systems to verify that the tests met TS surveillance requirements, UFSAR commitments, and in-service testing and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the SSCs were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement. This activity constituted five inspection samples.

In-Service Tests:

- 1-SR-3.5.1.6(CS 1), Unit 1 Core Spray Flow Rate Loop 1

Routine Surveillance Tests:

- 0-SR-3.8.1.1(C), Diesel Generator C Monthly Operability Test
- 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test
- 1/2/3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration
- 2-SR-3.5.3.3, RCIC System Rated Flow at Normal Operating Pressure

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

During the report period, the inspectors observed an Emergency Preparedness (EP) Severe Accident Management Guidelines (SAMG) training drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures on March 14, 2012. This drill was intended to identify any licensee weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room, Technical Support Center, and Operations Support Center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure, and licensee conformance with other applicable Emergency Plan Implementing Procedures. The inspectors also attended the post-drill critiques to compare any inspector-observed weaknesses with those identified by the licensee in order to verify whether the licensee was properly identifying EP related issues and entering them in to the CAP, as appropriate. This activity constituted one inspection sample.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

4OA1 Performance Indicator (PI) Verification

.1 Reactor Coolant System (RCS) Activity and RCS Leakage

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-2.2, Performance Indicator Program. The inspectors examined the licensee's PI data for the specific PIs listed below for the first through fourth quarters of 2011. 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 six inspection samples.

- Unit 1 RCS Activity
- Unit 1 RCS Leakage
- Unit 2 RCS Activity
- Unit 2 RCS Leakage
- Unit 3 RCS Activity
- Unit 3 RCS Leakage

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

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

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

(CARB) and PER Screening Committee (PSC) meetings. This activity constituted one inspection sample.

.2 Annual Follow-up of Selected Issues – Root Cause Report for Unit 1 FCV-74-52 Valve Failure

Inspection Scope

The inspectors reviewed the details surrounding PER 410394 which documented the failure of the Unit 1 Low Pressure Coolant Injection (LPCI) valve 74-52 on August 2, 2011, due to its Limitorque actuator failing. Specifically, the inspectors confirmed that this issue was classified as the highest priority 'A' level PER requiring a root cause report in accordance with licensee procedures. The inspectors evaluated the thoroughness of the root cause report, the corrective actions and corrective actions to prevent recurrence (CAPRs) associated with this PER with a specific focus on corrective action and CAPR extensions. Additionally, the inspectors evaluated the licensee's extent of condition review and observed the inspections and reassembly of five other LPCI injection valves. No additional valve failures were observed during these inspections. However during the inspection of the Unit 1 74-66 valve actuator, indications of abnormal wear were observed on specific parts of the clutch assembly. These parts were replaced and the valve tested satisfactorily before returning the valve to service. This activity constituted one inspection sample.

Findings and Observations

No findings were identified. The inspector observed weaknesses associated with management involvement in the CAP extension approval process.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Reports (LERs) 05000259/2011-009-00 and -01, As-Found Undervoltage Trip for the Reactor Protection System 1A1 Relay that Did Not Meet Acceptance Criteria During Several Surveillances

a. Inspection Scope

The inspectors reviewed LER 05000259/2011-009-00 dated December 5, 2011, the revised LER 05000259/2011-009-01 dated January 31, 2012, and PERs 413140 and 442914, including the associated root cause analysis, operability determination, and corrective action plans. On October 6, 2011, while performing an operability determination for the Unit 1 reactor protection system (RPS) 1A1 relay undervoltage trips, the licensee determined that the as-found undervoltage trip for the RPS 1A1 relay was less than the required acceptance criteria during several TS surveillances performed between April 2007 and August 2011. Six of the last nine surveillance test results were below the TS acceptance criteria. Therefore, based on performance history, the RPS 1A1 relay was determined to be inoperable from April 30, 2007 to October 5, 2011, when the relay was replaced. The licensee determined the root cause to be lack of specific instructions in the surveillance test program for past operability reviews when out of TS conditions were corrected during surveillances.

b. Findings

The enforcement aspects of this finding are discussed in Section 4OA7. These LERs are considered closed.

.2 (Closed) LERs 05000259/2010-003-00, -01 and -02, Failure of a Low Pressure Coolant Injection Flow Control Valve

a. Inspection Scope

The inspectors reviewed LER 05000259/2010-003-00 dated December 22, 2010, and the revised LERs 05000259/2010-003-01 and 05000259/2010-003-02 dated April 1, 2011 and February 10, 2012 respectively. The inspectors reviewed PERs 271338 and 369800 to validate the accuracy of the reported root causes and the corrective actions. On October 23, 2010, during a refueling outage for Unit 1, the licensee discovered that a Residual Heat Removal (RHR) Loop II low pressure coolant injection (LPCI) flow control valve (74-66) failed to open when attempting to establish shutdown cooling while in Mode 3, Hot Shutdown. Additionally, LER revisions 01 and 02 described a Part 21 report corresponding to the first root cause which was an undersized thread barrel due to a manufacturing defect. When subjected to a system differential pressure greater than the capacity of the reduced thread engagement, the valve skirt could separate from the disc. The licensee also discovered two additional root causes during the investigation of this event. The lack of requirements for verification of thread dimensions resulted in failure to identify the undersized thread barrel during reassembly of the new valve disc with the old valve skirt in 1983. Also, the mischaracterization of the active/passive safety function for valve 74-66 resulted in the inappropriate classification and removal from the Generic Letter (GL) 89-10 program. GL 89-10 describes a program to ensure valve motor-operator switch settings (torque, torque bypass, position limit, overload) for motor-operated valves (MOVs) in several specified systems are selected, set, and maintained so that the MOVs will operate under design-basis conditions for the life of the plant.

b. Findings

The enforcement aspects of this issue were discussed in NRC Inspection Report 05000259/2011-008. No additional findings were identified regarding the original or revised LERs. The NRC is conducting supplemental inspections in accordance with Inspection Procedure 95003, Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input. These LERs are considered closed.

.3 (Closed) LERs 05000259/2011-008-00, and -01, High Vibrations on High Pressure Coolant Injection Booster Pump Thrust Bearings.

a. Inspection Scope

The inspectors reviewed LER 05000259/2011-008-00 and LER 05000259/2011-008-01 dated September 19, 2011 and January 31, 2012, respectively. Inspectors reviewed PER's 408067 and 405165 related to this event. The HPCI booster pump exhibited increased vibration and the licensee made the initial conservative determination that the condition affected the mission time and operability of the HPCI system for an unknown period. The initial LER stated that a supplement was forthcoming following further analysis. The revised LER provided further event analysis including; event causes, a timeline of significant system events and, extent of condition which included verification of correct installation on other similar bearings. In addition, the licensee's analysis concluded that, given the as-found condition of the thrust bearings, the HPCI booster pump would not have been able to meet its mission time from May 20, 2011 until successful repairs on July 27, 2011.

b. Findings

One finding of significance related to the original LER 05000259/2011-008-00 was documented in IR 05000259/2011005 (see Licensee Identified Violations Section 4OA7). No additional findings were identified regarding the original or revised LER. These LERs are considered closed.

.4 Fire Event in the Turbine Building

a. Inspection Scope

On March 21, 2012, a fire occurred in the Unit 1 Turbine Building due to an overloaded 120 VAC circuit that was powering three submersible pumps. The inspectors performed an event follow up inspection of this fire to gather details surrounding this event. The inspectors reviewed the Main Control Room logs and interviewed Fire Operations and Operations on-shift personnel and verified no spurious alarms or spurious safety-related equipment operation. The inspectors performed a walkdown of the 557' elevation of the Turbine Building where the fire occurred to verify the extent of fire damage. It was determined that three extension cords were placed in series to achieve the length necessary to support the work. A ground fault circuit interrupter (GFCI) device was connected at the end of these extension cords, not at the electrical outlet. The three submersible pumps were plugged into the GFCI. Each submersible pump was rated for 9 amps. The 120 VAC outlet that the three submersible pumps were being powered from was rated for 20 amps. The licensee determined that the circuit breaker for this power outlet tripped four separate times and was reset each time until a fault occurred at the plugs connecting the two extension cords. These actions resulted in a small electrical fire. The licensee performed additional inspections of the plant and either removed extension cords where not in use or verified each extension cord had a GFCI as required. The inspectors performed walkdowns of additional plant areas for proper extension cord use.

b. Findings

One finding was identified.

Introduction: The NRC identified a Green NCV of Technical Specifications 5.4.1.d, Fire Protection Program implementation associated with the licensee's failure to report a fire in the Unit 1 Turbine Building to the main control room (MCR). Specifically, the failure to report a plant fire resulted in a failure of the MCR operators to implement Emergency Plan Implementing Procedure EPIP-17, Fire Emergency Response.

Description: On March 21, 2012, a plant worker discovered an electrical extension cord on fire on the 557' elevation of the Unit 1 Turbine Building. The worker contacted a member of Fire Operations who was in the vicinity performing inspections of fire protection equipment. The Fire Operations member immediately acted to put out the fire by unplugging the burning extension cord and extinguished the fire with a carbon dioxide (CO2) extinguisher. The turbine building auxiliary unit operator (AUO) was contacted to unplug the remaining extension cord from the power outlet. The Fire Operations Dispatch Report stated the fire was extinguished within 1 minute upon notification to a member of the Fire Operations staff. After the fire was extinguished, the Fire Operations member contacted the Unit 1 Unit Supervisor in an attempt to locate the Shift Manager. The Shift Manager was touring the Turbine Building around the same time and was notified of the fire. The Shift Manager went to the scene of the fire and concurred the fire was out and determined the scene where the fire occurred was safe. The Fire Operations Shift Captain was also notified of the fire and arrived at the scene to document the fire event and extension cord damage. The licensee determined that the cause of the extension cord fire was an overloaded circuit. Specifically, a ground fault circuit interrupter (CFGI) device was improperly connected at the end of these extension cords and three submersible pumps were plugged into the GFCI. Each submersible pump was rated for nine (9) amps. The 120VAC outlet that the three submersible pumps were being powered from was rated for twenty (20) amps. The licensee determined that the circuit breaker for this power outlet tripped four times and was reset each time until a fault occurred at the plugs connecting two extension cords and resulted in a small electrical fire.

The inspectors challenged the licensee's response to the fire and questioned why the personnel involved did not immediately report the fire to the MCR using the phone number for plant emergencies. The site's Plant Access Training (PAT000) instructed all plant personnel to report fires immediately by dialing 3911 on an installed plant phone and to alert others in the area. Emergency Plan Implementing Procedure EPIP-17, Fire Emergency Response, required the MCR to initiate the Fire Alarm bell, announce the fire location over the plant's public address system and notify the Fire Protection personnel and the Shift Manager of the fire. Because the MCR was not immediately notified of the fire in the Unit 1 Turbine Building, EPIP-17, was not entered, the Fire Alarm bell was not initiated, and the Fire Brigade was not contacted to respond to the fire. The licensee entered this event into their corrective action program as PER 527090.

Analysis: Failure to immediately report a fire in the Unit 1 Turbine Building was a performance deficiency. As a result, the MCR was not immediately notified of a plant fire and EPIP-17, Fire Emergency Response, was not implemented. The performance deficiency was determined to be more than minor because if left uncorrected, the failure to notify the MCR of plant fire events would have the potential to lead to a more significant safety concern. Specifically, emergency response procedures for plant fires would not be implemented and Fire Brigade response would be delayed. The finding was associated with the Initiating Events Cornerstone and initially characterized according to IMC 0609, Significance Determination Process (SDP), Attachment 4, Phase 1 - Initial Screening and Characterization of Findings. The results of this analysis required an evaluation in accordance with IMC 0609, Appendix F, Attachment 1, Part 1, Fire Protection SDP Phase 1 Worksheet. For the SDP Phase 1 evaluation a low degradation rating was assigned for this fire event because it was a small electrical fire with no combustible material within the vicinity of the fire. Additionally, the licensee had previously established one-hour roving fire watches in place throughout the plant to meet other requirements with specific training to immediately report fires to the MCR. The finding was determined to be of very low safety significance (Green). The cause of this finding was directly related to the cross cutting aspect of Procedural Compliance in the Work Practices component of the Human Performance area, because the licensee failed to recognize the requirement to immediately report a fire and enter the appropriate fire emergency response procedures [H.4(b)].

Enforcement: Technical Specifications 5.4.1.d, Fire Protection Program implementation requires in part that the licensee will establish, implement and maintain written procedures covering implementation of the Fire Protection Program. Contrary to this requirement, licensee personnel failed to implement EPIP-17, Fire Emergency Response, for a fire in the Unit 1 Turbine Building, on March 21, 2012. Following the event, plant staff performed additional inspections of plant areas and either removed electrical extension cords or ensured each cord had a required GFCI and was not overloaded. Expectations for plant employees discovering and responding to fires were reinforced by plant management. Because the violation was of very low safety significance and has been entered into the licensee's CAP as PER 527090, this violation is being treated as an NCV consistent with the NRC Enforcement Policy. This NCV is identified as NCV 05000259, 260, 296/2012002-02, Failure to Implement Fire Protection Program Procedures.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Operation of an Independent Spent Fuel Storage Installation (ISFSI) (60855)

a. Inspection Scope

The inspectors observed operations involving independent spent fuel storage installation-related activities, interviewed personnel, and reviewed licensee documentation to verify that the ISFSI-related programs and procedures fulfilled the commitments and requirements specified in the Safety Analysis Report (SAR); the Certificate of Compliance (CoC), including the technical specifications (TSs); and 10 CFR Part 72, including 10 CFR 72.48 evaluations and 10 CFR 72.212(b) evaluations for general licensed ISFSIs. In addition, the inspectors observed selected ISFSI-related activities and conducted independent evaluations to ensure that the licensee performed spent fuel loading and transport in a safe manner and in compliance with approved procedures.

The inspectors reviewed six 10 CFR 72.48 Screening Reviews for several ISFSI procedures and verified that all changes were consistent with the license and CoC, and did not reduce program effectiveness.

The inspectors attended a pre-job briefing and observed operations in the field including overall supervisory involvement, coordination, and oversight of ISFSI-related work activities. The inspectors observed lifting of a loaded HI-TRAC cask and the transfer of the multi-purpose canister (MPC) into the HI-STORM cask via the stack up configuration. The inspectors noted that the field supervisor maintained strict control of the work package and continually verified that procedural steps were followed and completed as required. The inspectors reviewed the fuel loading plan for MPC-0237 (the MPC being transported to the ISFSI pad during this inspection) and verified that the fuel assemblies identified were properly selected and loaded in accordance with characterization documents and approved procedures. The inspectors also reviewed the fuel loading plans for selected other MPCs which had been previously loaded and transported to the ISFSI pad and verified that the fuel assemblies identified were properly selected and loaded.

The inspectors verified that selected individuals had received the necessary training in accordance with approved procedures for their ISFSI-related job duties.

The inspectors reviewed a self-assessment report, two QA audits, and a benchmarking report conducted by the licensee to evaluate the effectiveness of the licensee's management oversight and QA assessments of ISFSI activities.

The inspectors reviewed the Dry Cask Radiological Work Permit, the As Low As Reasonably Achievable (ALARA) Planning Report, and dose estimates for the current ISFSI campaign. The inspectors noted that the ALARA plan was comprehensive with appropriate radiological controls established to minimize personnel exposures. The inspectors observed effective contamination control techniques and dose control measure implementation in the field. Radiological conditions were effectively communicated to individuals throughout the task. Radiological surveys of the loaded cask were obtained to ensure that radiation levels and contamination levels met the requirements of the CoC for safe storage of the HI-STORM cask at the ISFSI. The inspectors discussed the retention and maintenance of ISFSI-related records with station personnel and noted that appropriate arrangements had been made to maintain these records. The inspectors also reviewed the special nuclear material (SNM) inventory forms of SPP-5.8, Special Nuclear Material Control, for MPC-0234 and two others from previously loaded HI-STORM casks on the ISFSI pad.

The inspectors walked down the transfer path from the truck bay, where the MPC is loaded into the HI-STORM, to the ISFSI pad to verify that fire and explosive controls were being implemented in accordance with CoC surveillance requirements.

The inspectors determined that the licensee had established, maintained, and implemented adequate control of dry cask processing operations, including loading, transportation, and storage per approved procedures and that technical specification requirements and acceptance criteria as outlined in the Final Safety Analysis Report were followed appropriately. Records of spent fuel stored at the facility were properly maintained. The inspectors verified that changes to the design and operation were appropriately evaluated under 10 CFR 72.48. The inspectors determined that radiation protection controls were adequately established and implemented.

b. Findings

No findings were identified.

.3 (Closed) Unresolved Item (URI) 05000259, 260 and 296/2011003-03, Use of Inappropriately Qualified Methods to Evaluate Emergency Core Cooling During Accident Mitigation

a. Inspection Scope

The NRC inspection staff reviewed the licensee actions taken as a result of the failure to maintain an error in the ECCS Evaluation Model ECCS Evaluation Model described in EMF-2361(P)(A), “EXEM BWR-2000 ECCS Evaluation Model.”

b. Findings

This URI is considered closed with one NRC identified findings and one NRC identified Violation.

(1) Introduction: The NRC identified a Green non-cited violation of 10CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, for the licensee’s failure to ensure that the ECCS was satisfactorily designed

such that the maximum fuel element cladding temperature specified in 10 CFR 50.46(b) would not be exceeded.

Description: During discussions between the NRC staff, the fuel vendor, and the licensee starting in April 2010, the NRC staff questioned the appropriateness of the application of credit for spray cooling in the Units 2 and 3 ECCS evaluation and the effect non-single failure proof ADS would have on the ECCS evaluation model for the BFN units.

In a letter dated April 30, 2010 the licensee acknowledged the single failure issue and committed to modify the ADS to provide a single failure proof automatic initiation capability of 4 ADS valves. The licensee also outlined the compensatory measures intended to address the identified degraded/nonconforming condition. Additionally, the licensee committed to bring the ADS into compliance on Unit 3, during the spring 2012 outage and to Unit 2, during the spring 2013 outage.

In Inspection Report 0500259,260, 296/2011003, the inspectors reviewed Calculation ANP-2908(P), “Browns Ferry Units 1, 2, and 3 105% OLTP [loss of coolant accident] LOCA Break Spectrum Analysis.” The inspectors determined that the analysis, which used the ECCS Evaluation Model described in EMF-2361(P)(A) was not an adequate evaluation for application at Browns Ferry. The Browns Ferry ECCS evaluation was unique for two reasons: (1) in most BWR cases, the ADS was single failure-proof; however, for Browns Ferry it was not, and (2) the most severe postulated LOCA were those arising from small breaks, rather than a large break. Therefore, certain aspects of the approved evaluation model were not applicable to the unique plant configuration at Browns Ferry. The unique plant configuration was associated with an error made by the fuel vendor regarding the proper application of credit for spray cooling of fuel bundles during a small break LOCA. The NRC staff’s observation was documented by the licensee in PER 372764 on May 21, 2011. This error was only applicable to Units 2 and 3, as Unit 1 operated with a different fuel type.,

Because of the misapplication of spray cooling, the evaluation model described in EMF-2361(P)(A) was not entirely applicable to Units 2 and 3 while the ADS system design was considered to be non-single-failure-proof. The single failure issue involved a loss of the 250 volts- direct current (VDC) battery supplying power to Reactor Motor Operated Valve (RMOV) Board B. Both logic trains of automatic ADS initiation instrumentation were powered from RMOV Board B; consequently, loss of power to the board resulted in the loss of all automatic ADS function.

When the error was evaluated by the NRC staff consistent with Appendix K to 10 CFR 50 requirements, as illustrated in the Figure 6.19, of ANP-2908(P), the NRC staff determined that the fuel cladding temperature increase would continue until the time of rated core spray (CS) flow. At 500 seconds, the calculated peak cladding temperature would exceed 2200 degrees Fahrenheit, resulting in a high possibility of fuel failure.

The licensee entered the issue into the CAP as PER 372764 and instituted reactor protection system thermal limit compensatory measures on May 29, 2011.

Analysis: The inspectors determined that the licensee's failure to ensure that the ECCS was satisfactorily designed such that the maximum fuel element cladding temperature would not be exceeded was a performance deficiency. Specifically, the licensee failed to accurately maintain design control regarding single-failure assumptions for the ECCS and perform sufficiently bounding analyses to ensure that the calculated maximum fuel element cladding temperature of 2200 degrees Fahrenheit was not exceeded.. This performance deficiency was considered greater than minor because it was associated with the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents. Specifically, the failure to maintain the ADS single-failure proof, coupled with an ECCS modeling error, resulted in the failure of the design of the ECCS to ensure that the calculated maximum fuel element cladding temperature would not be exceeded in the event of a small break LOCA. The inspectors assessed the finding using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP), Attachment 4, and was determined to be of very low safety significance (Green), based on the remaining barriers to fission product release were unaffected.

The cause of this finding was directly related to the cross-cutting aspect of Issue Identification in the Corrective Action Program component of the Problem Identification and Resolution area because the licensee failed to completely, accurately, and in a timely manner identify the errors with the ECCS evaluation model [P.1.(a)].

Enforcement: 10 CFR Section 50.46 (a)(1)(i), requires, in part, that each boiling water nuclear power reactor be provided with an ECCS that is designed to ensure that the calculated cooling performance following postulated LOCAs does not exceed a calculated maximum fuel element cladding temperature of 2200 degrees Fahrenheit. Contrary to the above, from April 16, 2010 to May 29, 2011, the BFN Units 2 and 3 ECCS evaluation model, EMF-2361(P)(A), "EXEM BWR 2000 ECCS Evaluation Model, " the implementation and results for which are provided in ANP-2908(P), "Browns Ferry Units 1, 2, and 3 105% OLTP [Original Licensed Thermal Power] LOCA Break Spectrum Analysis," failed to ensure that the ECCS was satisfactorily designed such that the maximum fuel element cladding temperature of 2200F would not be exceeded in the event of a small break LOCA. On May 29, 2011, operating limitations were implemented to account for the error in calculations. Because the finding was determined to be of very low safety significance (Green) and has been entered into the licensee's CAP as PER 372764, this violation is being treated as an NCV consistent with the Enforcement Policy. This NCV is identified as NCV 05000260(296)/2012002-03, Failure to Ensure ECCS Design Calculation Does Not Exceed Maximum Clad Temperature.

(2) Introduction: The NRC identified a SL-IV NCV of 10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, for the licensee's failure to report a significant error discovered in their application of the ECCS model that affected the peak cladding temperature calculation.

Description: During discussions between the NRC staff, the fuel vendor, and the licensee starting in April 2010, the NRC staff questioned the appropriateness of the application of credit for spray cooling in the Units 2 and 3 ECCS evaluation, and the effect non-single failure proof ADS would have on the ECCS evaluation model for the BFN units.

In a letter dated April 30, 2010 the licensee acknowledged the single failure issue with ADS and indicated that the estimated effect of the change or error on peak clad temperature (PCT) was not significant (greater than 50 degrees Fahrenheit). TVA committed to modify the ADS to provide a single failure proof automatic initiation capability of 4 ADS valves. The licensee also outlined the compensatory measures intended to address the identified degraded/nonconforming condition. Subsequently, on June 30, 2011, TVA submitted the annual ECCS evaluation model report and indicated a minor change to the radiative heat transfer model which resulted in a minor change in PCT for Units 2 and 3. On October 7, 2011, TVA submitted a revised ECCS analysis in support of a Unit 1 fuel transition request. This analysis provided a methodology change to address the evaluation model error associated with spray cooling, which had been identified by the NRC staff, and for which the licensee implemented operating restrictions to ensure that the effects of the error would not cause the predicted PCTs at Units 2 and 3 to exceed 2200F.. This analysis was also applicable for current operating conditions for Units 2 and 3 and was not previously reported to the NRC. NRC review identified that the effect of the evaluation model error would have resulted in greater than a 50 degree increase in predicted PCT for Units 2 and 3.

On February 29, 2012, TVA initiated Service Request 514121 which recognized that a 30-day report for a significant change in peak clad temperature consistent with 10 CFR 50.46 had not been submitted. As of March 30, 2012, TVA had not submitted the required 30-day report for a significant change in peak clad temperature consistent with 10 CFR 50.46 which was identified on February 29, 2012. Following the end of the reporting period, TVA submitted the required report per 10 CFR 50.46 on April 18, 2012.

Analysis: The inspectors determined that the licensee's repeated failure to report changes or errors in the ECCS analyses was a performance deficiency. The inspectors reviewed this issue in accordance with IMC 0612, Appendix B, and determined the performance deficiency did not constitute a Finding, but the failure to report impacted the regulatory process and was subject to traditional enforcement consistent with the discussion for Block 7, Figure 2, Paragraph 2.a.v. The violation was determined to be more than minor per the NRC Enforcement Manual, Section 2.10.F, since the NRC has evidence that this failure to report has occurred repeatedly. This violation was determined to be a Severity Level IV violation based on section 6.9 of the NRC Enforcement Policy.

Enforcement: 10 CFR 50.46 (a)(3)(ii), requires for each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with 10 CFR 50.46 requirements.

Contrary to the above, the licensee failed to report each change or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation for Units 2 and 3. Specifically, from May 29, 2011 to April 18, 2012, the licensee failed to report a significant change in peak clad temperature associated with an error related to spray cooling to the NRC within 30 days, and include with the report a proposed schedule for providing reanalysis or taking other action as may be needed to show compliance. The licensee subsequently submitted the required report per 10 CFR 50.46. Because this violation was determined to be a Severity Level IV violation and was entered into the licensee's CAP as PER 531752, this violation is being treated as an NCV consistent with the Enforcement Policy. This NCV is identified as NCV 05000260(296)/2012002-04, Repeated Failure to Report ECCS Analyses Methodology Change or Errors.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On March 30, 2012, regional inspectors presented the inspection results specifically associated with ISFSI activities via telephone with Michael Durr, Director of Engineering, and other members of the licensee's staff.

On April 6, 2012, the resident inspectors presented the inspection results to Mr. Lang Hughes, Operations Manager, and other members of the licensee's staff, who acknowledged the findings.

On April 13, 2012, the resident inspectors presented the inspection results specifically associated with closure of unresolved item (URI) 2011-003-03, Use of Inappropriately Qualified Methods to Evaluate Emergency Core Cooling System During Accident Mitigation, to Mr. Lang Hughes, Operations Manager, and other members of the licensee's staff, who acknowledged the findings.

All proprietary information reviewed by the inspectors as part of routine inspection activities were properly controlled, and subsequently returned to the licensee or disposed of appropriately.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and was a violation of NRC requirements which met the criteria of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation.

- Unit 1 Technical Specification 3.3.8.2, Reactor Protection System (RPS) Electric Power Monitoring, required that, for each in-service RPS motor generator set or alternate power supply, two RPS electric power monitoring assemblies be operable in Modes 1, 2, and 3; and in Modes 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. With one electric power monitoring assembly inoperable for one or both in-service power supplies, the associated in-service power supply(s) were required to be removed from service in 72 hours or be in Mode 3 within 12 hours and in Mode 4 within 36 hours. In addition, TS 3.0.4 prohibited Mode changes with TS 3.3.8.2 not met. Contrary to this, on October 6, 2011, while performing an operability determination for the channel A RPS power monitoring system undervoltage trips, the licensee determined that the as-found undervoltage trip for the RPS 1A1 relay was less than the required TS acceptance criteria during multiple previous TS surveillances and that the RPS 1A1 relay was inoperable from April 30, 2007 to October 5, 2011. This TS violation was entered into the licensee's CAP as PERs 413140 and 442914. The finding was determined to be of very low safety significance because the finding does not represent an actual loss of the RPS safety function in that the remaining operable channel A RPS electric power monitoring assembly still provided protection to the RPS bus powered components under degraded voltage conditions.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
K. Polson, Site Vice President
S. Bono, Plant General Manager
D. Hughes, Operations Manager
M. Durr, Director of Engineering
J. Boyer, Acting Assistant Director of Engineering
B. Bruce, Acting Systems Engineering Manager
D. Carter, TVA QA Program Manager
J. Emens, Nuclear Site Licensing Manager
A. Feltman, Emergency Preparedness Manager
J. Ferguson, Radiation Protection Support Superintendent
M. Hydas, Project Manager, Sequoyah Dry Cask Project
S. Kelly, Work Control Manager
D. Kettering, Electrical Systems Engineering Manager
R. King, Design Engineering Manager
P. Summers, Director of Safety and Licensing
Z. Martin, Corporate Nuclear Fuels
B. McNutt, Shift Manager
R. Norris, Radiation Protection Manager
S. Norris, Engineering Supervisor
P. Parker, Site Security Manager
E. Quidley, EDG Project Manager
R. Kerving, Performance Improvement Manager
M. Rasmussen, Operations Superintendent
H. Smith, Fire Protection Supervisor
J. Underwood, Chemistry Manager
C. Vaughn, Operations Superintendent
S. Walton, Electrical Maintenance Superintendent
M. Wilson, Director of Training
A. Yarbrough, BOP System Engineering Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000259, 260, 296/2012002-01	NCV	Failure to Adequately Implement Impaired Fire Barrier and Detector Controls (Section 1R05)
05000259, 260, 296/2012002-02	NCV	Failure to Immediately Report a Plant Fire (Section 4OA3.4)
05000260, 296/2012002-03	NCV	Failure to Ensure ECCS Design Calculation Does Not Exceed Maximum Clad Temperature (Section 4OA5.3)
05000260, 296/2012002-04	NCV	Repeated Failure to Report ECCS Analyses Methodology Change or Errors (Section 4OA5.3)

Closed

05000259/2011-009-00	LER	As-Found Undervoltage Trip for the Reactor Protection System 1A1 Relay that Did Not Meet Acceptance Criteria During Several Surveillances (Section 4OA3.1)
05000259/2011-009-01	LER	As-Found Undervoltage Trip for the Reactor Protection System 1A1 Relay that Did Not Meet Acceptance Criteria During Several Surveillances (Section 4OA3.1)
05000259/2010-003-00	LER	Failure of a Low Pressure Coolant Injection Flow Control Valve (Section 4OA3.2)
05000259/2010-003-01	LER	Failure of a Low Pressure Coolant Injection Flow Control Valve (Section 4OA3.2)
05000259/2010-003-02	LER	Failure of a Low Pressure Coolant Injection Flow Control Valve (Section 4OA3.2)
05000259/2011-008-00	LER	High Vibrations on High Pressure Coolant Injection Booster Pump Thrust Bearings (Section 4OA3.3)
05000259/2011-008-01	LER	High Vibrations on High Pressure Coolant Injection Booster Pump Thrust Bearings (Section 4OA3.3)
05000259, 260, 296/2011-003-03	URI	Use of Inappropriately Qualified Methods to Evaluate the Emergency Core Cooling System

Discussed
None

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

0-OI-82, Standby Diesel Generator System, Rev. 122 (Attachments 2, 2A, 3, 3A, 4A)
1-OI-74, Residual Heat Removal System, Rev. 76
1-OI-74/ATT-1, RHR System Attachment 1 Valve Lineup Checklist, Rev. 77
1-OI-74/ATT-2, RHR System Attachment 2 Panel Lineup Checklist, Rev. 76
1-OI-74/ATT-3, RHR System Attachment 1 Electrical Lineup Checklist, Rev. 76
DWG 1-47E811-1, Flow Diagram RHR System, Rev. 37
UFSAR, Section 8.5, Standby AC Power Supply and Distribution, Amendment 24
0-OI-82, Standby Diesel Generator System, Rev. 125
0-OI-82/ATT-1C, Standby DG C Valve Lineup Checklist, Rev. 100
0-OI-82/ATT-2C, Standby DG C Panel Lineup Checklist, Rev. 100
0-OI-82/ATT-3C, Standby DG C Electrical Lineup Checklist, Rev. 100
0-OI-82/ATT-4C, Standby DG C Instrument Inspection Checklist, Rev. 102
0-OI-18, Fuel Oil System, Rev. 52
0-OI-18, Attachment 1, Valve Lineup Checklist, Rev. 49
0-OI-67, Emergency Equipment Cooling Water System, Rev. 94
0-OI-67/ATT-1, Attachment 1Valve Lineup Checklist Unit 0, Rev. 83
Emergency Diesel Generators System Health Report, Dated 10/1/2011 - 1/31/2012
Function 082-B Diesel Generator (a)(1) Plan, Rev 1 Effective Date 06/29/2010
MSPI Margin to WHITE, Report dated February 2012
DCN 69454, DG Turbocharger Lube Oil System Modification, Rev. A
PMTI-69454-STG003, Post Modification Test Instruction – DCN 69454 – C Diesel Lube Oil Modification, Rev. 6
PER 207876, Diesel Generator System - Re-Status to Maintenance Rule A1 Status
PER 295257, DG D fast start failure
PER 367040, 1C EDG's need the exhaust piping insulation replaced
PER 394449, EDG Turbo Oil Pressure Gauges
PER 362395, Oil leak resulting in emergency shutdown of C DG
PER 405925, Fuel Transfer Pump High Amps
PER 407393, C Diesel Generator Fuel Priming Pump high amps
PER 418097, DG Air Start System PMs were cancelled
PER 424092, DG Lube Oil Usage
PER 445811, QA Finding – Untimely Resolution of EDG Tech Spec
PER 479400, Priority 2 WO's Not Being Worked
PER 486972, Drilling Hole for Conduit Support for Conduit
PER 488920, Adverse Trend in the Quality of Modification Work Orders
PER 490734, Damaged Motor Leads 0-MTR-082-0007C, AC Soakback Pump Motor
PER 490986, Post Modification Testing of the Diesel Generator Lube Oil Modification DCN 69454
PER 493017, Design Discrepancy DCN 69454 - DG Lube Oil Mod
PER 494419, C DG Lube Oil System Immersion Heater Control Problem
PER 495020, Tubing clearances
PER 496375, C DG Outboard Bearing Vibration Levels still high after recent outage work
PER 496599, DG C Soakback Pump (AC) not running
PER 511014, C DG AC Lube Oil Soakback Pump returned to service without corrective action following failure.

PER 517290, Diesel Generator "C" soakback oil pressure reading out of spec
PER 520468, Perform Generator Alignment on DG C
PER 527139, Diesel Generator Governor Tightness Check
PER 527208, Re-impact DCN 69454 stage 3
WO 111123549, WO to correct minor oil leak on Diesel Generator Governor (PER 229785-001)
WO 112248022, 1C EDG's need the exhaust piping insulation replaced
WO 112486945, C Diesel Generator Fuel Priming Pump high amps
WO 112548187, Replace turbocharger oil pressure gage for DG C with the gage specified in PER Action 394449-001
WO 112812546, DCN 69454 Stage 3 Install New Circulating and Soakback Pumps and Associated Piping
WO 112812556, DCN 69454 STAGE 3. Install New Lube Oil System Vent Lines And Gallery Fill Tubing And Supports
WO 113132405, Design Discrepancy DCN 69454 - DG Lube Oil Mod
WO 113160417, BFN-0-PMP-082-0007C [D/G C SOAKBACK PUMP (AC)] NOT RUNNING.
WO 113191583, C Diesel Generator Engine Control Cabinet local alarms did not have an audible alarm when tested
WO 113256179, C DG AC Lube Oil Soakback Pump returned to service without corrective action following failure.
WO 113292348, DG C Soakback Oil Pressure Reading Out of Spec
WO 113310390, WORK ORDER ONLY - Perform generator alignment on DG C
DWG 0-47E861-3, Flow & Control Diagram Diesel Starting Air Sys. Diesel Generator C, Rev. 12
DWG 0-47E861-7, Flow Diagram – Cooling System & Lubricating Oil Sys. Standby Diesel Generator C, Rev. 14
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UFSAR, 8.4 Normal Auxiliary Power System, Amendment 23
0-OI-57A, Switchyard and 4160V AC Electrical System, Rev. 141
0-OI-57A/ATT-1, Attachment 1, Valve Lineup Checklist, Rev. 134
0-OI-57A/ATT-2, Attachment 2, Panel Lineup Checklist, Rev. 133
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Operations Logs dated 3/26/12 to 3/28/12
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Section 1R05: Fire Protection

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Fire Protection Report, Volume 1, Fire Hazards Analysis Units1/2/3, Fire Zone 2-5, Rev. 11
Fire Protection Report, Volume 2, Section IV.6, Pre-Plan Nos. RX2-621 and RX2-639, Unit 2 Reactor Building Elevations 621'-3" and 639'-0", Rev. 8,

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NPG-SPP-18.4.6, Control of Fire Protection Impairments, Rev. 0
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0-AOI-26-1, Fire Response, Rev. 13
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Fire Protection Report Vol. 2, Rev. 48
Fire Protection Report Vol. 1, Fire Hazards Analysis, Rev. 11
Fire Protection Report Vol. 2, Rev. 48
0-SI-4.11.G.1.a, Visual Inspection of Fire Rated Barriers (Floors, Walls & Ceiling), Rev. 20
PER 511557, Drain Slot
PER 514109, Repeat Drain Slot from Closed PER 511557
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Fire Protection Report Vol. 2, Rev. 48
0-47W216-56, Fire Area Compartmentation and Zone Drawings Panel 593 and 586, Rev. 06
SR 514773, Combustibles left on top of electric board room, U2 Rx bldg. El 593
SR 516388, Cable trays contain excess material
0-SSI-25-1, Intake Pumping Station EL 550, Cable Tunnel to Fire Door 440, RHRSW Pump Room B, RHRSW Pump Room D, Rev. 4
1-SI-4.11.A.1, Annual Smoke Detector Functional Test, Rev. 26
1-SI-4.11.A.3, Monthly Functional Test of Non-Supervised Alarm Circuits, Rev. 33
Branch Technical Position ASB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants, Rev. 1
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Drawing 0-47W600-788, Mechanical Fire Protection System Electrical Cable Tunnel EL 550 Location Plan, Rev. 3
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FPIP-12-3394, Beam Detectors Disabled
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PER 526380, Water in the Intake Tunnel
SR 525798, FP Equipment Availability
SR 526246, Fire Watch Requirements for Intake Cable Tunnel
SR 529011, Interim Actions for FPIPs
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SR 529360, Surveillance 1-SI-4.11.A.3 Exceeded Grace Period for Panel 297
SR 529361, Fire Impairment 11-2888 Did Not Identify the LCO Associated with G.1.a.2
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0-GOI-300-5, Environmentally Qualified Doors, Rev. 12
0-SR-3.6.4.1.1, Secondary Containment Equipment Hatches and Access Doors Position Verification, Rev. 11
0-SI-4.11.G.2, Semiannual Fire Door Inspection, Rev. 24
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0-SI-4.11.G.2.b, Fire Door Inspection, Rev. 18
1-OI-74, Residual Heat Removal System, Rev. 76
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2-ARP-9-3D, Panel 2-9-3, 2-XA-55-3D, Window 23, (Spare), Rev. 28
3-ARP-9-3D, Panel 3-9-3, 3-XA-55-3D, Window 23, RX BLDG Water Tight Door Open, Rev. 28
2-EOI-3, Unit 2 Secondary Containment Control, Rev. 11
BFN 50-C-7105, General Design Criteria for Pipe Rupture, Internal Missiles, Internal Flooding, and Vibration Qualification of Piping, Rev. 11
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Browns Ferry Nuclear Plant (BFN) – Moderate Energy Line Break (MELB) Flooding Evaluation, Appendix A, Reactor Building Moderate Energy Line Break Evaluation
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Drawing 0-46E454-22, Door & Hardware Schedule Appendix R, Rev. 5
Drawing 0-47E225-137, Required Reactor Building Door Positions to Ensure EQ Profiles are not Exceeded, Rev. 5
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MPI-0-260-DRS001, Inspection and Maintenance of Doors, Rev. 40
PER 336818, Evaluate BFN Internal Flooding Strategies
PER 340505, Verify BFN Capability to Mitigate Flooding Events
PER 344531, Use McGuire Procedure to Evaluate BFN Internal Flooding Strategies
PER 349897, DG Exterior Drains Not Seismic
PER 349898, DG Portable Bulkhead Doors and Sink Piping Not Seismic
PER 367611, Walkdown of 0-AOI-57-1A, LOOP/SBO
PER 481471, 2" Diameter Hole and 24" Deep in U1/2 DG Outer Wall
PER 498970, Water Intrusion Trends

Section 1R11: Licensed Operator Requalification

OPL177.041, H2 Supply Alarm, HPCI Pressure Switch Failure, Condenser Tube Leak, Fuel Failure, Main Steam Line leak, Uniolable RCIC Steam Line Break, HPCI Failure, 2 Area Rad Levels Above Max Safe., Rev. 8
OPDP-1, Conduct of Operations, Rev. 22

Section 1R12: Maintenance Effectiveness

NPG-SPP-03.4, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting – 10CFR50.65, Rev. 0

0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting – 10CFR50.65, Rev. 37
Unit 1, 2 and 3 Function 73-B (a)(1) Plan, Rev. 0
PER 152914, Maintenance Rule (a)(1) Status for U2 HPCI
PER 226507, U2 HPCI Elevated Casing Temperatures > 190 degrees
PER 228565, Repeat Issues with HPCI Turbine Steam Supply Valve Leakage
PER 379134, U3 HPCI Elevated Casing Temperature
PER 382507, U1 HPCI - Elevated Casing Temperature
PER 413752, U2 HPCI - Elevated Casing Temperature
PER 436575Continued Leakage Issues with HPCI Steam Admission Valves
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MCI-0-000-GTV002, Double Disc, Pressure Seal Gate Valves, Rev. 3
MCI-0-000-GTV002, Double Disc, Pressure Seal Gate Valves, Rev. 4
WO 111148387, U2 HPCI – WO for Replacing Diaphragm on 2-PCV-073-0018C
WO 113239029, Pressure transmitter sensing line is not firmly attached to support
WO 112507920, Check for correct thrust bearing installation in 3-PMP-073-0029 HPCI booster pump
PER 65935, Unit 3 HPCI turbine casing temperatures
PER 84090, U3 HPCI
PER 116989, U2 Elevated HPCI Casing Temperature
PER 144253, Unit 2 HPCI oil sample had high moisture
PER 146171, Unit 1 HPCI moisture in the oil
PER 147128, 1-FCV-73-16 failure
PER 152914, Maintenance Rule A1 Status for U2 HPCI
PER 175435, U3 HPCI - Elevated Turbine Casing Temperatures
PER 224634, U3 HPCI Main Pump Seal Leak Needs Repaired
PER 226507, Elevated HPCI Casing Temperatures > 190 F
PER 228565, Repeat issues with HPCI Turbine Steam Supply Valve Leakage
PER 235338, HPCI isolation during performance of 2-SR-3.3.6.1.6(3)
PER 239313, Unit 2 HPCI inadvertently isolated
PER 246674, U2 HPCI - Packing Leak on 2-FCV-073-0035 and old WO canceled
PER 372659, U1 Gland Seal Condenser leaking
PER 402417, U1 HPCI testable check valve entering (a)(1) status

PER 408067, Unit 1 HPCI Booster Pump outboard bearings found installed incorrectly
PER 413752, U2 HPCI - Elevated Turbine Casing Temperature
PER 436575, Contiued leakage issues with HPCI Steam Admission valves
PER 436575, Continued leakage issues with HPCI Steam Admission valves
PER 503340, Handwheel for 1-ISIV-073-0031B is missing
PER 503343, Handwheel for 1-ISIV-073-0031 is missing
PER 505472, 0-E-4707 does not show as built configuration for Unit 1 HPCI skid grouting connections
PER 507867, Flex Conduit Seal unsecured
PER 507893, Missing HPCI Conduit Box Cover
PER 509604, Poor Housekeeping Practices on U1 HPCI
PER 509645, C-Zone on South side of U2 HPCI does not meet general cleanliness standards
PER 509650, Seismic Hazards on both U2 HPCI Balconies
PER 510120, Fire Hazards on U2 HPCI Steam Supply Line
PER 510424, Failure to respond to a potential threat to U2 HPCI
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0-SR-3.8.1.1(C), Diesel Generator C Monthly Operability Test, Rev. 40
1/2-ARP-9-23C, Panel 9-23-8 Window 15, Diesel Gen C Prot Relay Operation, Rev. 18
1/2-ARP-9-23C, Panel 9-23-8 Window 30, 4160V Bkr 1722, 1812, or 1814 Overload, Rev. 18
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SR 510119, DG Instantaneous Power Ratings Unclear in Design Output Documents (TS, FSAR, Design Criteria)
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PER 488920, Adverse Trend in the Quality of Modification Work Orders
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WO 112815809, DCN 69454 Stage 3 Install DC Control Panel
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PER 401732, During performance of WO 09-718159-000, step 4.4.6 was not satisfied.
PER 403936, Work Order needed to repair 3C Emergency Diesel Generator Rotor (Reference PER 401732)
PER 405695, WO ONLY Needed to reperform AC Pole Drop and Impedence Test on the D/G
PER 470404, Interim actions associated with Functional Evaluations not scheduled
PER 480886, 3C EDG Operability requires evaluation
WO 09-718159-000, BFN has historically not performed polarization Index (PI), and Bearing resistance testing on the EDGs
WO 112472092, Work Order needed to repair 3C Emergency Diesel Generator Rotor (Reference PER 401732)
WO 112486450, WO ONLY Needed to reperform AC Pole Drop and Impedence Test on the DG
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3-47E812-2, Flow Diagram High Pressure Coolant Injection System, Rev. 06
PER 50188, HPCI Turbine Stop Valve 3-FCV-073018 did not open as required
PER521782, Develop preventative maintenance for spring replacement
SR 521663, Inspect the trip tappet assembly for FME
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PER 486778, Spacing violations on anchor bolts
PER 488622, ESI Non-Conformance Report for stubshaft, Item BTT336K, NCR# 6626
WO 112815845, DCN 69454 Stage 3 Install Components
PER 494429, Two Relays damaged during installation
WO 112884807, DCN 69454 STAGE 3 Perform Post-Cal Testing
WO 112812600, DCN 69454 STAGE 3 Flush Lube Oil System After Modifications
WO 09-720111-002, Pressure test new DG Lube Oil Cooler per MCI-0-082-CLR002 In support of DCN 69454
WO 112884901, DCN 69454 STAGE 3 Perform Post Modification Testing For the Mechanical Portion of DCN 69454 Stage 3
WO 09-710573-000, Implement Diesel Generator Turbocharger Lube Oil System Modifications (DCN 69454)
WO 112812546, DCN 69454 Stage 3 Install New Circulating and Soakback Pumps and Associated Piping
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PER 490986, Post Modification Testing of the Diesel Generator Lube Oil Modification DCN 69454
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PER 494419, C DG Lube Oil System Immersion Heater Control Problem
PER 495020, Tubing clearances
PER 495662, PMTI-69454-STG003 failed step 6.19.[11]
ESI Statement Regarding Soakback Pressure, dated 1/24/2012
PER 496599, D/G C Soakback Pump (AC) not running
PER 496375, C DG Outboard Bearing Vibration Levels still high after recent outage work
PER 495677, Vibration Device Connection Method
PER 492664, Temporary Diesel Generator fuel hose damage
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WO 113160417, BFN-0-PMP-082-0007C [D/G C SOAKBACK PUMP (AC)] NOT RUNNING
System Design Criteria BFN-50-7200E, Browns Ferry Nuclear Plant 4KV AC Auxiliary Power System, Rev. 10
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SR 483054, Battery Chargers for the Temporary Diesels need 480V with a Neutral
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PER 490734, Damaged Motor Leads 0-MTR-082-0007C, AC Soakback Pump Motor
PER 486778, Spacing violations on anchor bolts
PER 488622, ESI Non-Conformance Report for stubshaft, Item BTT336K, NCR# 6626
WO 112815845, DCN 69454 Stage 3 Install Components
PER 494429, Two Relays damaged during installation
WO 112884807, DCN 69454 STAGE 3 Perform Post-Cal Testing
WO 112812600, DCN 69454 STAGE 3 Flush Lube Oil System After Modifications

WO 09-720111-002, Pressure test new DG Lube Oil Cooler per MCI-0-082-CLR002 In support of DCN 69454
WO 112884901, DCN 69454 STAGE 3 Perform Post Modification Testing For the Mechanical Portion of DCN 69454 Stage 3
WO 09-710573-000, Implement Diesel Generator Turbocharger Lube Oil System Modifications (DCN 69454)
WO 112812487, DCN 69454, Stage 3 Perform Platform Modifications
WO 112812546, DCN 69454 Stage 3 Install New Circulating and Soakback Pumps and Associated Piping
WO 112812552, DCN 69454 Stage 3 Install New Pipe Supports for New Circulating and Soakback Pump Piping
WO 112812556, DCN 69454 Stage 3. Install New Lube Oil System Vent Lines and Gallery Fill Tubing and Supports
WO 112815809, DCN 69454 Stage 3 Install Dc Control Panel
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PER 493017, Design Discrepancy DCN 69454 - DG Lube Oil Mod
PER 494419, C DG Lube Oil System Immersion Heater Control Problem
PER 493979, Kilowatts meter in control room ~ 200 kW higher than local indications
WO 113137443, Kilowatts meter in control room ~ 200 kW higher than local indications
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PER 495684, Functional Evaluation Tracking
PER 489573, 120V Heater Plug to TDG Breaker Found Unplugged
PER 496375, C DG Outboard Bearing Vibration Levels still high after recent outage work
PER 495677, Vibration Device Connection Method
PER 492664, Temporary Diesel Generator fuel hose damage
SR 505944, DG C Generator Watts recorded as 3650 KW in error
1-SR-3.5.1.6(RHR I), Quarterly RHR System Rated Flow Test Loop 1
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WO 112806263, Disassemble, clean heat exchanger, perform eddy current test
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0-TI-389, Raw Water Fouling and Corrosion Control, Rev. 15
SR 503288, WO 112902966 in COMP status without all required PMTs signed off
SR 506761, Operational risk review sheets not filled out correctly
NPG-SPP-06.1, Work Order Process, Rev. 0
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SR 511098, PMT not documented for 2B RFPT PDS replacement
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PER 515342, Clarification to NPG-SPP-06.1
NPG-SPP-06.1, Work Order Process, Rev. 0
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3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure, Rev. 62
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SR 511118 – Evaluate flow test of Aux steam supply check valve for PMT
SR 506530 – PMT for closed WO not signed off as expected
SR 511316 – Items not completed during WO close out process
1-SR-3.6.1.3.5(HPCI), HPCI System Motor Operated Valve Operability, Rev. 9
1-SR-3.3.3.1.4(G), Verification Of Remote Position Indicators For HPCI System Valves, Rev. 2
WO 112347240, Perform MOVATS testing on 1-MVOP-73-16 to support ODMI (PER 382507)
WO 112347282, Internal Repair of 1-FCV-073-0016, HPCI Steam Admission Valve
WO 112692321, Required Maintenance to Support Work of 1-FCV-073-0016
WO 112713129, Perform MOVATS testing on 1-MVOP-73-16 to support ODMI (PER 382507)
WO 113262364, Contingency WO, Inspect Internal Components
WO 113233526, WO for Internal Repair of 1-FCV-073-0016 HPCI Steam Admission Valve
WO 113265725, 1-SR-3.3.3.1.4(G) – Verif. of Remote Position Indicators for HPCI Sys. Valves
WO 113265732, HPCI System Motor Operated Valve Operability
WO 113268407, U1 PHCI Casing Temperature Trending Up
WO 113268636, Trouble Shoot 1-FCV-073-0016 for Seat Leakage by Manual Adjustment
WO 113272032, HPCI System Motor Operated Valve Operability
WO 113272282, Replace Coupling Insert on 1-PMP-073-0047
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0-TI-364, ASME Section XI System Pressure Tests, Rev. 15
PER 382507, U1 HPCI - Elevated Turbine Casing Temperature
PER 511607, U1 HPCI Steam Admission Valve
PER 513173, maximum set point seating thrust and torque was exceeded on 1-FCV-073-0016

<u>Section 1R22: Surveillance Testing</u>
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0-SR-3.8.1.1(C), Diesel Generator C Monthly Operability Test, Rev. 42
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0-OI-82, Diesel Generator Operating Logs, dated 1/24/2012
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WO 113146639, D/G C Idle Speed
PER 486972, Hole Drilled in C D/G Day Tank
PER 495275, D/G C Idle Speed
PER 495677, Vibration Device Connection Method
PER 495684, Functional Evaluation Tracking
PER 496375, C DG Outboard Bearing Vibration Levels still high after recent outage work

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0-OI-82, Standby Diesel Generator System, Rev. 125
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WO 112815368, EDG 'D' Monthly Test
WO 113212635, D DG Lube Oil Consumption Test
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1/2/3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Revs. 0, 5, and 4 respectively
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PER 338020, Data Reporting for NRC Performance Indicator "Maximum RCS Identified Leakage" Changed
PER 346011, Incorrect Max RCS Identified Leakage Reporting

SR 517276, Drywell Equipment Drain Sump Level Abnormal, Lowering Level With No Pump Operation
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NPG-SPP-03.1, Corrective Action Program, Rev. 02
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PER 131365, Out of Tolerance Time Delay Relay
PER 151812, RPS Circuit Protector Failed Acceptance Criteria
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PER 271338, Failure of the Unit 1 FCV-74-66 valve
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UNS S17400, AK Steel 17-4 PH Stainless Steel Product Data Sheet
Root Cause PER 408067, Unit 1 HPCI Booster Pump Bearing Failure
LER 05000259/2011-008-00, High Vibrations on High Pressure Coolant Injection Booster Pump Thrust Bearings
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PER 378921, High Vibrations identified on U1 HPCI Main Pump & Booster Pump
PER 405165, HPCI Vibration Point in alert per PDM VIB meter
PER 408067, Unit 1 HPCI Booster Pump outboard bearings found installed incorrectly
PER 496420, Discrepancies Noted in Root Cause Prepared for PER 408067 concerning HPCI Booster Pump

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CR 502588, Criticality Alarm of Monitor #860287
CR 502592, BFN Engineering Cannot Support Needed 72.48 Evaluations/Screenings
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PER 204419, Repetitive weakness in BFN Site implementation of ISFSI Regulatory Requirements
PER 376258, Root Cause Analysis Report, Rev 2, Nine Multi-Purpose Canisters/HI-STORMs were loaded with spent fuel without being receipt inspected
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Browns Ferry Radiological Survey # 050411-24
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Personnel Training Records
Special Nuclear Material Inventory Form for twenty-five HI-STORM casks on ISFSI Pad, dated 9/21/2010
Letter L17 091023 801, Transmitting ISFSI QA Audit BFA0901 of October 2 to October, 2009
Letter L17 110630 800, Transmitting ISFSI QA Audit SSA 1110 of May 16 to June 1, 2011
Letter L32 100426 800, Documenting the Selection of Unit 1 and Unit 2 fuel Assemblies for the Fourth Dry Cask Storage Campaign, dated April 26, 2010
Letter L32 120106 802, documenting the selection of Unit 3 2 fuel assemblies for the fifth Dry Cask Storage campaign, dated January 6, 2012
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Completed Technical Instruction 0-TI-508, Fuel Assembly Inspection Prior to MPC Loading, Revision 2, for MPC 0243 (Campaign No. 4)
Completed Technical Instruction 0-TI-508, Fuel Assembly Inspection Prior to MPC Loading, Revision 3, for MPC 0234 (Campaign No. 5 – the current campaign)
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L17 091023801, QA-TVA Nuclear Power Group BF Audit No: BFA0901, BFN ISFSI, October 27, 2009

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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