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

May 13, 2011

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2011002, 05000260/2011002, 05000296/2011002, 07200052/2011002 AND NOTICE OF VIOLATION

Dear Mr. Krich:

On March 31, 2011, 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 11, 2011, with Mr. Keith Polson 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.

Based on the results of this inspection, the NRC has determined that a Severity Level IV violation of NRC requirements occurred. The violation was evaluated in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's Web site at (http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html). This violation is cited in the enclosed Notice of Violation and the circumstances surrounding it are described in detail in the subject inspection report. The violation is being cited in the Notice because it involved the repetitive failure to adequately control transient combustible materials inside the Independent Spent Fuel Storage Installation (ISFSI) area and within close proximity of the dry casks loaded with spent fuel. This violation is being cited because the criterion specified in Section 2.3.2.a.3 of the NRC Enforcement Policy for a non-cited violation was not met. This criterion was not met because the violation was repetitive and identified by the NRC. Two previous violations were identified by the NRC and were documented in NRC Inspection Reports 07200052/2010002 and 07200052/2010003 respectively.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Furthermore, this report documents two NRC-identified findings and two self-revealing findings that were evaluated under the risk significance determination process as having very low safety significance (Green). Three of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and they were entered into your corrective action program, the NRC is treating these findings as NCVs consistent with the Enforcement Policy. Additionally, four licensee-identified violations which were determined to be of very low safety significance are listed in this report. If you contest any violation, 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 Senior Resident Inspector at Browns Ferry Nuclear Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the Senior Resident Inspector at 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, 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),accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal, privacy or proprietary information so that it can be made available to the public without redaction.

Sincerely,

/RA/

Eugene F. Guthrie, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosures:
1. Notice of Violation
2. NRC Integrated Inspection Report 05000259/2011002, 05000260/2011002, 05000296/2011002 and 07200052/2011002

cc w/encl. (See page 3)

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cc w/encl. (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE

ADAMS: ☐ Yes ACCESSION NUMBER:___ML111330758_______ ☐ SUNSI REVIEW COMPLETE

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS
SIGNATURE	Via email	Via email	Via email	EFG /RA for/	Via email	Via email	Via email
NAME	TRoss	CStancil	PNiebaum	LPressley	MCoursey	SWalker	PHiggins
DATE	05/13/2011	05/13/2011	05/13/2011	05/13/2011	05/11/2011	05/11/2011	05/11/2011
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
OFFICE	RII:DRS	RII:DRP	RII:DRP	RII:DRP	RII:DRP		
SIGNATURE	Via email	Via email	Via email	CRK /RA/	EFG /RA/		
NAME	RPatterson	CFletcher	RHamilton	CKontz	EGuthrie		
DATE	05/11/2011	05/11/2011	05/11/2011	05/13/2011	05/13/2011		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\DRPII\RPB6\BROWNS FERRY\REPORTS\2011\BF 002\BF INSPECTION REPORT-11-02.DOCX

cc w/encl:
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Letter to R. M. Krich from Eugene Guthrie dated May 13, 2011

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2011002, 05000260/2011002, 05000296/2011002, 07200052/2011002 AND NOTICE OF VIOLATION

Distribution w/encl:
C. Evans, RII
L. Douglas, RII
OE Mail
RIDSNRRDIRS
PUBLIC
RidsNrrPMBrownsFerry Resource

NOTICE OF VIOLATION

Tennessee Valley Authority	Docket No. 07200052
Browns Ferry Nuclear Plant	License No. DPR-68

During an NRC inspection conducted on February 3, 2011, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 72.212, Conditions of general license issued under §72.210, section (b)(9) stated, in part, that the licensee shall "Conduct activities related to storage of spent fuel under this general license only in accordance with written procedures." Procedure NPG-SPP-18.4.7, Control of Transient Combustibles, stated that requirements and controls for handling and use of transient combustibles associated with the Independent Spent Fuel Storage Installation (ISFSI)/Dry Cask Storage Pad were contained within drawings 0-47E201-1 and 0-47E201-2. In particular, Item 11 of drawing 0-47E201-2, ISFSI Fire Hazards Analysis Compensatory Actions, established a 30 feet limit for wooden structures in proximity to a loaded HI-STORM cask stored on the ISFSI Pad. Also, General Operating Instruction (GOI) 0-GOI-300-1/ATT-12, Outside Operator Round Log, established instructions for daily inspections to ensure the ISFSI Pad and exclusion area were clear of flammable material such as wood; and if the area was not clear, to report the results to the Unit 3 unit supervisor (US) for evaluation of acceptability in accordance with drawing 0-47E201-2.
>
> Contrary to the above, seven storage cradles, multiple storage pallets and storage devices or cribbing, all constructed of wood products, were discovered on or near the ISFSI pad located between approximately 10 and 20 feet from a loaded HI-STORM cask on February 3, 2011. For approximately one week, these combustible materials had been located well within the 150 feet exclusion zone prescribed by 0-GOI-300-1/ATT-12, and the 30 feet limit of drawing 0-47E201-2. The outside operator also failed to report the transient combustible material to the Unit 3 US as required per 0-GOI-300-1/ATT-12, and no fire hazards evaluation according to drawing 0-47E201-2 was performed. The licensee promptly removed the transient combustible material from the ISFSI Pad exclusion area, initiated PER 318694, and performed a post-evaluation of the transient combustible material. (Section 4OA5.2)

This is a Severity Level IV violation.

Pursuant to the provisions of 10 CFR 2.201, the Tennessee Valley Authority is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region II, and a copy to the NRC Senior Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation " and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this 13th day of May, 2011.

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2011002, 05000260/2011002, 05000296/2011002, 07200052/2011002
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, 2011 through March 31, 2011
Inspectors:	T. Ross, Senior Resident Inspector C. Stancil, Resident Inspector P. Niebaum, Resident Inspector L. Pressley, Resident Inspector C. Fletcher, Senior Reactor Inspector (1R07) M. Coursey, Reactor Inspector (1R08) S. Walker, Senior Reactor Inspector (1R17) P. Higgins, Senior Reactor Inspector (1R17) R. Patterson, Reactor Inspector (1R17) R. Hamilton, Sr. Health Physicist (2RS1, 4OA1, 4OA5) A. Nielsen, Sr. Health Physicist (2RS3, 2RS4) C. Dykes, Health Physicist (2RS2)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure 2

SUMMARY OF FINDINGS

IR 05000259/2011002, 05000260/2011002, 05000296/2011002, 07200052/2011002; 01/01/2011 – 03/31/2011; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Refueling and Other Outage Activities, Identification and Resolution of Problems, Followup of Events.

The report covered a three month period of inspection by the resident inspectors, three senior reactor inspectors, two reactor inspectors and a senior health physicist from Region II. One severity level IV cited violation (VIO), three non-cited violations (NCV) and one self-revealing finding (FIN) were identified. The significance of most findings is identified by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); the cross-cutting aspect was 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: Initiating Events

- Green. A self-revealing non-cited violation of Technical Specifications (TS) 5.4.1.a was identified for the licensee's failure to adequately implement operations instruction 2-OI-74, Residual Heat Removal System, to ensure the reactor cavity draindown flow path was isolated prior to suppression pool draindown. On March 25, 2011, Operations personnel inadvertently left a Residual Heat Removal (RHR) system drain valve in the open position which led to an uncontrolled draindown of the reactor pressure vessel (RPV) coolant to the suppression pool. Operators immediately identified the RPV level decrease and restored the valve lineup and water level. The licensee's immediate corrective actions re-emphasized adherence to log keeping and turnover requirements; instituted shift manager challenges on activities that impact key safety functions including assessments of procedures, plant configuration, turnover information, and pre-job briefs of personnel roles and responsibilities; and, for those same activities, instituted peer checks, marked up drawings, and supervisory review of completed field copies of procedures. This issue was entered into the licensee's corrective action program as problem evaluation report (PER) 344533.

  This finding was considered more than minor because it was associated with the Human Performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, a mispositioned RHR drain valve resulted in a loss of control of the RPV water level. This finding was determined to be of very low safety significance (Green) according to Inspection Manual Chapter (IMC) 0609, Appendix G, Shutdown Operations, because the inadvertent loss in excess of 2 feet (approximately 40 inches) of reactor coolant inventory represented a loss of inventory control. Using IMC 0609, Appendix G, Attachment 3, "Phase 2 Significance Determination Process Template for BWR During Shutdown," a Senior Reactor Analyst performed an analysis and determined the loss of inventory event was of very low risk significance (Green) due in part to automatic functions being available to isolate and mitigate the leak had it continued and remained undetected/uncorrected by the operators. The cause of this finding was directly related to the cross-cutting aspect of Work Activity Coordination in the Work Control component of the Human Performance area, because inadequate documentation and communication of plant system configuration by the control room operators resulted in a mispositioned valve and loss of RPV water level [H.3.(b)]. (Section 1R20.1)

- Green. An NRC identified non-cited violation of 10 CFR 50 Appendix B, Criteria XVI, Corrective Action, was identified for the licensee's failure to correct a condition adverse to quality related to Unit 3 primary containment isolation system (PCIS) logic relays exceeding their in-service life expectancy. Specifically, the licensee failed to replace numerous Unit 3 PCIS CR120A relays prior to exceeding their vendor's recommended service lifetime. The licensee has entered this issue into their corrective action program as problem evaluation report (PER) 348160.

  This finding was determined to be more than minor because it was associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the frequency of those events that upset plant stability and challenge critical safety functions during power operations. Specifically, a relay failure could cause a reactor scram, engineered safeguards (ESF) actuation, and/or Group 1, 2, 3, or 6, primary containment isolation. The significance of the finding was evaluated using Phase 1 of the significance determination process in accordance with the Inspection Manual Chapter (IMC) 0609 Attachment 4, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross cutting aspect of Appropriate Corrective Actions in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee failed to implement adequate corrective actions as part of PER 220336 to replace or extend the service life of the Unit 3 PCIS CR120A relays prior to exceeding their recommended service lifetime [P.1(d)]. (Section 4OA2.2)

  Green. A self-revealing finding (FIN) was identified for the licensee's failure to adequately evaluate and take the required actions established by site standards to address an adverse system performance trend that had degraded below acceptable levels associated with the main generator exciter air coolers. Specifically, the licensee failed to identify that main generator exciter air cooler differential temperatures exceeded the licensee-defined limit of 10F, and did not initiate a PER as required by the licensee's procedural guidance, Nuclear Engineering Department Procedure (NEDP) -20, Conduct of the Engineering Organization, Section 3.1, System Performance Monitoring. Subsequent licensee corrective actions included installing vents on the exciter air coolers to minimize air binding, establishing a process and frequency for venting the exciter air coolers, and increasing engineering supervisory oversight of the system monitoring process. The licensee captured this issue in the corrective action program as PER 301505.

This finding is greater than minor because it is associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability. Specifically, the finding resulted in a Unit 3 manual reactor scram due to elevated main turbine bearing vibrations caused by excessive main generator exciter air cooler differential temperatures. The significance of the finding was evaluated using Phase 1 of the significance determination process in accordance with the Inspection Manual Chapter (IMC) 0609 Attachment 4, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting aspect of Corrective Action Program Implementation in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee failed to identify the adverse trend of excessive differential temperatures between the exciter air coolers in a timely manner and enter it into the corrective action program. [P.1(a)]. (Section 4OA3.2)

Cornerstone: Barrier Integrity

- Green. An NRC identified non-cited violation of Technical Specifications (TS) 5.5.2, Primary Coolant Sources Outside Containment was identified for the licensee's failure to establish, implement, and maintain an adequate program for minimizing primary coolant leaks from systems (i.e., Core Spray, Residual Heat Removal, High Pressure Coolant Injection, and Reactor Core Isolation Cooling) outside containment, that could contain highly radioactive fluids during a serious transient or accident, to levels as low as practicable. The licensee's corrective actions included identification, evaluation, and prioritization of all known primary coolant leaks outside containment; and development of a new program in accordance with 0-TI-578, Minimizing Primary Coolant Sources Outside Containment. This finding was entered into the licensee's corrective action program as problem evaluation report (PER) 317464.

  This finding was determined to be more than minor because if left uncorrected it could have led to a more significant safety concern. Specifically, the licensee's failure to effectively minimize and monitor primary coolant leakage outside containment could have resulted in increased main control room exposure and/or offsite dose during an accident due to excessive radioactive fission product releases into secondary containment. The finding was determined to be of very low safety significance (Green) according to IMC 0609, Appendix H, Containment Integrity Significance Determination Process, Section 6.0, Type B Findings, because the primary coolant leak rate into secondary containment was a small fraction of the leakage assumed in the design basis accident (DBA) safety analyses. The cause of this finding was directly related to the cross-cutting aspect Complete and Accurate Procedures in the Resources component of the Human Performance area because the licensee's existing procedures were inadequate and incomplete for addressing the program requirements of TS 5.5.2 [H.2.(c)]. (Section 4OA2.5)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full Rated Thermal Power (RTP) for most of the report period except for one planned downpower. On March 5, 2011, a planned downpower to 75 percent RTP was conducted to support a control rod sequence exchange and scram time testing. Unit 1 returned to full RTP later that same day.

Unit 2 operated at essentially full RTP during the report period except for two planned downpowers and a refueling outage (RFO). On January 7, 2011, a planned downpower to 80 percent RTP was conducted to support a control rod pattern adjustment and the unit returned to full RTP on January 8, 2011. On February 4, 2011, a planned downpower to 80 percent RTP was conducted to support a control rod pattern adjustment and the unit returned to full RTP on February 5, 2011. On February 26, 2011, the unit was shutdown for refueling outage U2R16 and remained shutdown through the end of the report period.

Unit 3 operated at essentially full RTP the entire report period except for a startup and power ascension from a forced outage and two planned downpowers. Unit 3 began the report period shutdown as a result of a manual reactor scram due to high vibration on the main generator exciter bearings. The unit was restarted on January 5, 2011, and returned to full RTP on January 8. A related control rod pattern adjustment was conducted on January 8 and 10, 2011, with return to full RTP on January 11. On February 11, a planned downpower to 95 percent RTP was conducted to support repairs to the 3C1 reactor feedwater heater. The unit returned to full RTP the same day. On March 11, 2011, a planned downpower to 80 percent RTP was conducted for a control rod sequence exchange and scram time testing. The unit returned to full RTP on March 12, 2011.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Impending Adverse Weather Conditions

a. Inspection Scope

On January 10th a severe winter storm affected the southeast and in particular northern Alabama, which resulted in significant accumulation of snow and ice, and severe cold temperatures at the site during the entire week. The inspectors reviewed the licensee’s overall preparations/protection for the expected weather conditions and observed the licensee’s implementation of general operating instruction 0-GOI-200-1, Freeze Protection Inspection, and NPG-SPP-10.14, Freeze Protection. The inspectors witnessed the licensee’s execution of freeze protection of vulnerable areas and buildings inside and outside the power block. The inspectors verified the completion of the freeze protection checklists for multiple days.

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.

- 2A Standby Liquid Control (SLC) System
- Unit 1 Core Spray (CS) System - Division II
- Unit 2 Core Spray (CS) System - Division II

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 3 High Pressure Coolant Injection (HPCI) system, using the applicable P&ID flow diagrams, 3-47E812-1 and 3-47E812-2, along with the relevant operating instruction, 3-OI-73, High Pressure Coolant Injection System, 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 five fire areas (FA) and fire zones (FZ) listed below. Selected FAs/FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure NPG-SPP-18.4.6. Furthermore, the inspectors reviewed applicable portions of the Site Fire Hazards Analysis Volumes 1 and 2 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 5 inspection samples.

- Unit 1/2/3 Intake Structure Cable Tunnel (FA-25)
- Unit 1 Reactor Building, EL 519 through 565, from column line R1 to 10 ft. east of column line R4 (FZ 1-1)
- Unit 1 Reactor Building, EL 519 through 565, from column line R7 to 10 ft. west of column line R4 and Southeast Stairway/Elevator Enclosure between columns R6 to R7 at EL 593, 621 and 639 (FZ 1-2)
- Intake Structure (FA-25)
- Residual Heat Removal Service Water (RHRSW) and Emergency Equipment Cooling Water (EECW) Pump Rooms, and Intake Structure deck (FA-25)

b. Findings

No findings were identified.

.2 Annual Fire Brigade Drill:

a. Inspection Scope

On February 23, 2011, the inspectors witnessed an unannounced fire drill in the Unit 1 Control Building in the Main Bank No. 1 Battery Room. The inspectors assessed fire alarm effectiveness; response time for notifying and assembling the fire brigade; the selection, placement, and use of fire fighting 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

a. Inspection Scope

The inspectors performed a walk down of the Raw Cooling Water (RCW) system in the Reactor and Turbine Buildings to identify appropriate system isolation valves. The inspectors also performed walk downs of risk-significant areas, susceptible systems and equipment, including the Unit 1, 2 and 3 RHR, CS pump rooms, HPCI pump room, and Under-Torus area to review flood-significant features such as area level switches, room sumps and sump pumps, flood protection door seals, conduit seals and instrument racks that might be subjected to flood conditions. The inspectors reviewed plant design features and measures intended to protect the plant and its safety-related equipment from internal flooding events, as described in the site's licensing basis documents. Plant procedures for mitigating flooding events were also reviewed to verify that licensee actions were consistent with the plant's design basis assumptions. Specific documents reviewed are listed in the attachment.

The inspectors also reviewed a sampling of the licensee's corrective action documents with respect to flood-related items to verify that problems were being identified and corrected. Furthermore, the inspectors reviewed selected preventive maintenance procedures, work orders, and surveillance procedures to verify that actions were completed in accordance with design basis documents.

b. Findings

No findings were identified.

1R07 Triennial Heat Sink Performance

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillance tests, calculations, performance test results, and inspection results associated with the 2A Residual Heat Removal heat exchanger (RHR HX) and the 3C Diesel Generator heat exchanger (DG HX). These heat exchangers were chosen based on their risk significance in the licensee's probabilistic safety analysis, their important safety-related mitigating system support functions, and their relatively low safety margin.
For the 2A RHR HX and the 3C DG HX, the inspectors verified whether testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs

were adequate to ensure proper heat transfer. This was accomplished by reviewing whether the test method used was consistent with accepted industry practices, or equivalent; the test conditions were consistent with the selected methodology; the test acceptance criteria were consistent with the design basis values; and results of heat exchanger performance testing were acceptable. The inspectors also verified whether the test results appropriately considered differences between testing conditions and design conditions; the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values; and whether test results considered test instrument inaccuracies and differences.

The inspectors reviewed the methods and results of heat exchanger performance inspections. The inspectors also verified whether the methods used to inspect and clean heat exchangers were consistent with as-found conditions and expected degradation trends, and industry standards. Furthermore, the inspectors verified the licensee's inspection and cleaning activities had established acceptance criteria consistent with industry standards; and the as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable.

The inspectors whether the condition and operation of the 2A RHR and 3C DG heat exchangers were consistent with design assumptions in heat transfer calculations and as described in the UFSAR. This included determining whether the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors also verified whether the licensee evaluated the potential for water hammer and established adequate controls and operational limits to prevent heat exchanger degradation due to excessive flow induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

In addition, the inspectors reviewed problem evaluation reports (PER) related to the heat exchangers/coolers, and heat sink performance issues to determine whether the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment.

These inspection activities constituted two heat sink inspection samples as defined in IP 71111.07.

b. Findings

No findings were identified.

1R08 Unit 2 - Inservice Inspection (ISI) Activities (IP 71111.08B)

a. Inspection Scope

From March 14-18, 2011, the inspectors observed and reviewed the implementation of the licensee's In-service Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary and risk-significant piping boundaries of Browns Ferry Unit 2 during the spring 2011 refueling outage i.e., U2R16). The inspectors' activities consisted of an on-site review of nondestructive examination (NDE) and welding activities to evaluate compliance with Technical Specifications and the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections XI and V (Code of record: 2001 Edition through the 2003 Addenda), for Class 1, 2, and 3 systems; and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code Section XI acceptance standards. For Browns Ferry Unit 2, this was the last outage in the third period of the third 10-year ISI inspection interval. The inspectors also reviewed a sample of inspection activities associated with components that are outside the scope of ASME Section XI requirements which are performed in accordance with commitments to follow industry guidance documents, such as the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

Piping Systems ISI. - The inspectors reviewed NDE activities, both by direct observation and record review, specifically including examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, and calibration reports for compliance to requirements of ASME Section V, ASME Section XI, BWRVIP documents, and other industry standards for the following examinations:

Visual Testing (VT)
- VT-3 Examination of RHR-2-R-266 support
- VT-3 Examination of RHR-2-H-32 rigid hanger
- VT-3 Examination of RHR-2-H-371 support

Ultrasonic Testing (UT)
- UT Examination of DSCS-2-14 weld pipe to elbow for Core Spray System
- UT Examination of DSCS-2-03 weld valve to penetration for Core Spray System
- UT Examination of DSCS-2-01 weld valve to pipe for Core Spray System
- UT Examination of DSCS-2-02 weld elbow to valve for Core Spray System

The inspectors conducted a Unit 2 containment walk-down of multiple drywell elevations to assess, in general, the material condition of structures, systems, and components, including leaks from bolted connections, coating integrity, cleanliness, hangers and supports, etc.

The inspectors also reviewed welding activities from the current outage for the following Class 1 and 2 components:

- Skirt to Disc weld 2-FCV-074-0066

The inspectors completed a review of ISI-related problems that were identified by the licensee and entered into the corrective action program(CAP). The inspectors reviewed these corrective action documents to confirm that the licensee had appropriately described the scope of the problems, and had implemented appropriate corrective actions. The inspectors' review included confirmation that the licensee had an adequate threshold for identifying issues. Through interviews with licensee staff and review of corrective action documents, the inspectors evaluated the licensee's threshold for identifying lessons learned from industry issues related to ASME Section XI. The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment.

b. Findings

No findings were identified.

.2 Reactor Vessel Internal Inspections

a. Inspection Scope

The inspectors reviewed the following NDE activities associated with the inspection of Reactor Vessel internal components (Boiling Water Reactors Vessel Internals Project):

Visual Testing (VT)

- Jet Pump assemblies on shroud and vessel side
- Core shroud support weld H8.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program

.1 Operating Experience Smart Sample FY 2010-02, Sample Selections for Reviewing Licensed Operator Examinations and Training Conducted on the Plant Referenced Simulator

a. Inspection Scope

On January 27, 2011, the inspectors witnessed the licensed operator requalification (LOR) as-left simulator evaluation per Unit 2 Simulator Exercise Guide OPL 177.073, "Power Reduction, Recirculation Pump Trip, Reactor Power Oscillations, and ATWS with MSIVs Open," of an onshift Operations crew that had received remedial training. Then on February 7, 2011, the inspectors observed as-found simulator evaluation per Unit 2 Simulator Exercise Guide OPL 177.076, "Security Event, Loss of 480V RMOV Board 2B, Suppression Pool Leak, ATWS, and Exceed PSP," of an onshift Operations crew. Both of these simulator exercises were inspected in accordance with the guidance of IP 71111.11, Licensed Operator Requalification Program, and Operating Experience Smart Sample (OPESS) FY 2010-02.

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

- Clarity, formality, and effectiveness of communications between crew members, and with support organizations outside the control room
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, verification, and response to multiple alarms and changing instrument readings
- Manual operator actions due to failures of automatic functions
- Correct use and implementation of multiple, concurrent Operating Instructions (OI), Annunciator Response Procedures (ARP), Abnormal Operating Instructions (AOIs), and Emergency Operating Instructions (EOIs), including applicable appendices
- 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 (US) and Shift Manager (SM) of available control room and field operator resources

The inspectors attended the post-examination critiques 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 portions of the simulator for 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).

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

.1 Routine

a. Inspection Scope

The inspectors examined two specific equipment issues listed below for 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) Charging 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 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 3.1, 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.

- 480VAC Shutdown Boards, Reactor Motor Operated Valves (RMOV) Boards, Diesel Aux Boards, and Load Shedding Functions performance criteria development
- Revised Control Rod Drive (CRD) Pump 10-point 10 CFR 50.65 (a)(1) plan

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 reviewed four maintenance risk assessments, and actions taken to plan and/or control work activities to effectively manage and minimize risk. The inspectors verified that risk assessments and applicable risk management actions (RMA) were conducted as required by 10 CFR 50.65(a)(4) and 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 adequacy of the licensee's risk assessments, implementation of RMAs, and plant configuration.

- 3ED 4160 volt (4KV) Shutdown Board, 3ED Emergency Diesel Generator (EDG), Standby Gas Treatment (SBGT) C, and D1 RHRSW Pump Emergent Maintenance with C3 EECW and D2 RHRSW Pumps OOS
- 2C RHR Heat Exchanger, 2AD Low Pressure Coolant Injection (LPCI) Motor Generator (MG) set, C and G Plant Control Air (PCA) Compressors, and the B EDG OOS

- Unit 2 Cycle 16 Refueling Outage Risk Assessment Report
- Unit 1/2 4KV Shutdown Board, 2A 480V Shutdown Board, and Unit 1/2 RHR Cross-Tie Valves OOS

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the seven 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.

- EDG Fuel and Oil Filters In-service Beyond Manufacturers Recommendations (PER 305861)
- 3B EDG (PER 242991), 3C EDG (PER 244412), and A EDG (PER 246551) High Vibrations
- Units 1, 2, and 3: Non-Conforming Conditions for the RHR System I and II Outboard Isolation Valves (PER 303097)
- Units 1, 2, and 3: Drywell Temperature Analysis for Station Blackout (SBO) Non-Conservative (PERs 248261 and 248262)
- Unit 3: Drywell Pressure Instrument Slope Non-Conforming (PER 242068)
- 2C RHR Heat Exchanger Partition Plate Corrosion (PER 314747)
- Unit 3: 3ED 4KV Shutdown Board Normal Feeder Breaker 52STA Switch Linkage Failure Resulting in a Loss of Automatic Start Capability for 3D RHR, 3D CS, and D1 RHRSW Pumps (PER 322640)

b. Findings

One finding was identified. The enforcement aspects of this finding are discussed in Section 4OA7.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to verify that the licensee had appropriately considered the conditions under which changes to the facility, UFSAR, or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for nine changes and additional information, such as drawings, calculations, supporting analyses, the UFSAR, and TS to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The nine evaluations reviewed are listed in the attached List of Documents Reviewed.

The inspectors reviewed samples of changes for which the licensee had determined that evaluations were not required, to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10CFR50.59. The 15 "screened out" changes reviewed are listed in the attached List of Documents Reviewed.

The inspectors evaluated engineering design change packages for six material, component, and design based modifications to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. The six modifications and the affected cornerstones were as follows:

- DCN 67324, RTO/S Modify Feedwater Control System Software For Extended Power Uprate Impact, Rev. A
- DCN 69409, Electro-Hydraulic Control Software Upgrade, Rev. A
- DCN 69901, Perform Joint Owners Group (JOG) Updates for Valve 1-FCV-75-25, Rev. A
- DCN 69332, Replacement Motor has Higher HP Affects Single Line/TOL Size, Rev. A
- DCN 66315, Modify Main Steam Isolation Valve Internal Configuration As Needed For Extended Power Uprate, Rev. A
- DCN 69448, Replace valves 1-FCV-071-0025 And 1-FCV-071-0034, Rev. A

Documents reviewed included procedures, engineering calculations, modification design and implementation packages, work orders, site drawings, corrective action documents, applicable sections of the living UFSAR, supporting analyses, TS, and design basis information. The inspectors additionally reviewed test documentation to ensure adequacy in scope and conclusion. The inspectors review was also intended to verify that all appropriate details were incorporated in licensing and design basis documents and associated plant procedures.

The inspectors also reviewed selected corrective action documents associated with modifications and screening/evaluation issues to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated and tracked to completion.

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the five post-maintenance tests (PMT) listed below to verify that procedures and test activities confirmed SSC operability and functional capability following maintenance. The inspectors reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors verified that PMT activities were conducted in accordance with applicable WO instructions, or procedural requirements, including NPG-SPP-6.3, Pre-/Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors reviewed problems associated with PMTs that were identified and entered into the CAP.

- Unit 3: PMT for Anticipated Transient Without Scram (ATWS)-Recirculation Pump Trip (RPT) Automatic Trip Unit 3-LS-3-58C1, Reactor Low Water Level, Replacement Per WO 111841543 and 3-SR-3.3.5.1.2(ATU B), Core and Containment Cooling Systems Analog Trip Unit Functional Test
- Unit 1/2: PMT for B EDG Lube oil filter, Fuel filter replacement and Heat exchanger Cleaning Per 0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, following
- C3 EECW Pump Shaft and Packing Replacement PMT Per 3-SI-4.5.C.1(2), EECW Pump Operation, and WO 111481416
- Unit 2: PMT 2-SR-3.5.1.6(RHR I-COMP) RHR Loop I Comprehensive Pump Test
- Unit 2: PMT for 2-FCV-74-67, Residual Heat Removal Loop II Inboard Injection Valve Joint Owners Group Repairs Per WOs 09-727665-000 and -001, 10676572, 110689184, and 110689220, and various Surveillance and Maintenance Procedures

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 2 Scheduled Refueling Outage (U2R16)

a. Inspection Scope

From February 26 through March 31, the inspectors examined critical outage activities associated with the U2R16 refueling outage to verify that they were conducted in accordance with TS, applicable operating procedures, and the licensee's outage risk assessment and management plans. Some of the more significant inspection activities conducted by the inspectors were as follows:

U2R16 Outage Risk Assessment

Prior to the Unit 2 scheduled U2R16 refueling outage that began on February 26, the inspectors met with outage risk assessment team members and reviewed the Outage Risk Assessment Report to verify that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing an outage plan that assured the necessary levels of defense-in-depth of safety functions were maintained. The inspectors also reviewed the daily U2R16 Refueling Outage Reports, including the Outage Risk Assessment Management (ORAM) Safety Function Status, and regularly attended the licensee's outage status meetings. These reviews were compared to the requirements in licensee procedure NPG-SPP-07.2, Outage Management. These reviews were also done to verify that for identified high risk significant conditions, due to equipment availability, severe weather and/or system configurations, that contingency measures were identified and incorporated into the overall outage and contingency response plan. Furthermore, the inspectors frequently discussed risk conditions and designated protected equipment with Operations and outage management personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process

The inspectors witnessed the shutdown and cooldown of Unit 2 on February 25 and 26, 2011,in accordance with licensee procedures OPDP-1, Conduct of Operations; 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; and 2-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 2-OI-74, Residual Heat Removal System; 2-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Alternate Decay Heat Removal System Failures. The inspectors also conducted main control room and plant walkdowns of the RHR (shutdown cooling mode) and Alternate Decay Heat Removal (ADHR) systems and components to verify proper alignment and system operation. There were no planned evolutions that resulted in an increased outage risk condition of "Orange" for shutdown cooling. However, inspectors verified that plant conditions and systems identified as “protected” in the risk mitigation strategy were available. In addition, the inspectors reviewed controls implemented to ensure that outage work was not impacting the ability of operators to operate spent fuel pool cooling, RHR (shutdown cooling), and/or ADHR systems.

Critical Outage Activities

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

- Walked down selected safety-related equipment clearance 2-074-0006A for Tagout 2-TO-2011-0003 for Division I of the RHR system
- Walked down selected safety-related equipment clearance 2-225-0001A for Tagout 2-TO-2011-0003 for the 2A 480V Unit Board.
- Verified RCS inventory controls, especially during evolutions involving operations with the potential to drain the reactor vessel (OPDRV) controlled per 2-POI-200.5
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) and TS compliance during the various shutdown modes of operation, and mode transitions
- Observed and examined implementation of reactivity controls
- Reviewed overall control of containment penetrations and primary containment integrity
- Examined foreign material exclusion controls particularly in proximity to and around the reactor cavity, equipment pit, and spent fuel pool
- Routine tours of the main control room, reactor building, refueling floor and drywell
- Verified no waivers pursuant to 10 CFR 26.207 were issued during the U2R16 RFO

Reactor Vessel Disassembly and Refueling Activities

The inspectors witnessed selected activities associated with reactor vessel disassembly, and reactor cavity flood-up and drain down in accordance with 2-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, on numerous occasions, the inspectors witnessed fuel handling operations during the Unit 2 reactor core fuel shuffles performed in accordance with TS and applicable operating procedures, such as 0-GOI-100-3A, Refueling Operations (In Vessel Operations); 0-GOI-100-3B, Operations in the Spent Fuel Pool Only; and 0-GOI-100-3C, Fuel Movement Operations During Refueling. The inspectors verified specific fuel movements as delineated by the Fuel Assembly Transfer Forms (FATF). Furthermore, on March 15, 2011, the inspectors independently reviewed the U2C17 Beginning of Cycle (BOC) core reload video used to verify fuel assembly location, orientation, and lack of foreign material in accordance with 0-GOI-100-3C, Attachment 6, Core Verification.

Drywell Closeout

On March 23, 2011, the inspectors conducted a closeout inspection of the Unit 2 Suppression Chamber to verify it met the requirements of licensee procedure 2-GOI-200-2, Primary Containment Initial Entry and Closeout, Section 5.4, Torus Closeout. Additionally, on April 2 and 4, the inspectors reviewed the licensee's conduct of 2-GOI-200-2, Section 5.3, Drywell Closeout, and performed an independent detailed closeout inspection of the Unit 2 drywell.

Pre- Restart Activities

The inspectors specifically conducted the following:

- Witnessed portions of heatup and pressurization, and depressurization, of the Unit 2 reactor pressure vessel in-service test accordance with 2-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor pressure Vessel and Associated Piping; and reviewed 2-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring; and 2-SR-3.4.9.1(2), Reactor Vessel Shell Temperature and Reactor Coolant Pressure Monitoring During In-Service Hydrostatic or Leak Testing
- Attended Plant Oversight Review Committee (PORC) meeting for Unit 2 restart on March 31, 2011

Corrective Action Program

The inspectors reviewed PERs generated during U2R16 and attended management review committee meetings to verify that initiation thresholds, priorities, mode holds, operability concerns and significance levels were adequately addressed. Resolution and implementation of corrective actions of several PERs were also reviewed for completeness.

b. Findings

One finding was identified.

Introduction: A self-revealing Green noncited violation (NCV) of TS 5.4.1.a, was identified for the licensee's failure to adhere to operations department procedure (OPDP)-1, Conduct of Operations, to adequately implement operations instruction (OI) 2-OI-74, Residual Heat Removal System, to ensure the reactor cavity draindown flow path was isolated prior to commencing a suppression pool draindown which resulted in an uncontrolled loss of Unit 2 reactor vessel level.

Description: On March 25, 2011, while aligning the Unit 2 RHR system to reduce suppression pool chamber water level, the control room operators noticed that reactor pressure vessel (RPV) water level began lowering unexpectedly with a corresponding rise in suppression pool water level. The operators immediately took action to restore water level in the RPV by injecting with the condensate storage and supply (CS&S) system via the CS system, and securing the suppression pool drain down alignment. The Unit 2 RPV water level had lowered approximately 36 inches from the RPV flange during the transient, and partially uncovered the main steam (MS) dryer, before it was secured. Unit 2 was in Mode 5 with reactor reassembly in progress (i.e., RPV closure head being reinstalled) as part of the planned recovery from refueling operations. However, with the MS dryer partially uncovered, and with the RPV closure head removed, a significant change in radiation levels occurred on the refuel floor. The initial measured dose rate prior to the transient was 10 millirem per hour (mr/hr) at the handrails of the refuel floor-to-reactor cavity interface. The measured dose rate at the lowest point of the RPV water level transient was 150 mr/hr at the handrails on the refuel floor. Calculated dose rates inside the reactor cavity were 8000 mr/hr and 13,500 mr/hr at the RPV flange stud bolts.

Following the event, the licensee promptly performed an RHR system valve alignment verification and determined that the normally locked closed RHR Pump A Suction Drain valve (2-DRV-74-525A) was inadvertently left open. At the time of the transient, Division I of RHR was in the shutdown cooling (SDC) mode of operation with 2C RHR Pump in service and 2A RHR Pump in standby. Both divisions of CS were available for RPV injection and Division II of RHR was OOS for planned outage maintenance. With the 2-DRV-74-525A valve open, and the Loop I RHR system in shutdown cooling mode, the subsequent alignment of the RHR Drain Pump A to lower suppression pool water level created a flow path that allowed the RPV to gravity drain directly to the suppression pool.

Further investigation by the licensee, of activities leading up to the transient, determined that several opportunities had existed for operators to identify the abnormal RHR system configuration (i.e., DRV-74-525A open). On March 24, 2011, the night shift operators had aligned RHR Drain Pump A to the suction of the 2C RHR Pump in preparation to reduce suppression pool water level. This activity was documented in the shift logs. However, the normal dayshift operators missed the shift turnover with night shift due to participation in just-in-time training (JIT). Relief operators, not normally associated with the Unit 2 outage, failed to recognize this information in the logs. Later that same day, the normal day shift crew returned from training and then aligned RHR Drain Pump A to the suction of 2A RHR Pump to commence reactor cavity draining. This activity was not documented in the logs. Then again, the normal night shift operators missed shift turnover due to participation in JIT. Although the relief operators, not normally associated with Unit 2, apparently knew of the RHR Drain Pump A alignment to both 2A and 2C RHR Pumps, they did not document nor include the information in their turnover to the normal crew operators returning from training. The night shift operators completed the reactor cavity draindown early on March 25, and secured the RHR Drain Pump A flow path to the suction of 2C RHR Pump. But this was also not documented in the logs. In the following turnover to day shift, the oncoming operators were informed that the reactor cavity draindown flow paths were secured. Unbeknownst to the day shift operators, the 2-DRV-74-525A valve had been left open while they commenced alignment of the RHR Drain Pump A to drain the suppression pool.

The licensee initiated PER 344533 to determine the root cause of the uncontrolled loss of RPV level. As part of their immediate corrective actions, the licensee re-emphasized adherence to shift log documentation and turnover requirements; instituted shift manager challenges on activities that impact key safety functions including assessments of procedures, plant configuration, turnover information, and pre-job briefs of personnel roles and responsibilities; and, for those same activities, instituted peer checks, marked up drawings, and supervisory review of completed field copies of procedures.

Analysis: The inspectors determined that the licensee's failure to maintain configuration control of RHR system status between multiple operating shifts was a performance deficiency which resulted in an uncontrolled Unit 2 RPV draindown to the suppression pool. This performance deficiency was considered greater than minor because it was associated with the Initiating Events Cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the mispositioned RHR drain valve resulted in a loss of control of the RPV water level. The inspectors determined the finding was potentially greater than very low safety significance because the inadvertent loss in excess of 2 feet (approximately 40 inches) of reactor coolant inventory represented a loss of inventory control. Using IMC 0609, Appendix G, Attachment 3, "Phase 2 Significance Determination Process Template for BWR During Shutdown," the Senior Reactor Analyst performed an analysis. The analysis determined the loss of inventory event was of very low risk significance (i.e., Green), due in part to automatic functions being available to isolate and mitigate the leak had it continued and remained undetected/uncorrected by the operators. Furthermore, the analysis determined that the operators had valid indication of RCS level, actions could be taken to isolate the leak in less than half½ the time before a loss of RHR, and the drain path could be isolated by at least one valve.

The cause of this finding was directly related to the cross-cutting aspect of Work Activity Coordination in the Work Control component of the Human Performance area, because inadequate documentation and communication of plant system configuration by the control room operators resulted in a mispositioned valve and loss of RPV water level [H.3.(b)].

Enforcement: Unit 2 Technical Specification 5.4.1.a. required that written procedures recommended in RG 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained. Procedures for operation of the safety related systems for vessel draining, shutdown cooling, and ECCS were specifically listed as recommended procedures by Sections 4.a, 4.e, and 4.h of Regulatory Guide 1.33, Appendix A. Unit 2 operating instruction, 2-OI-74, Residual Heat Removal System, Section 8.36, Reactor Vessel Draindown Assist/Level Control to the Main Condenser or Radwaste using RHR Drain System, specified only one of two RHR pump suction drain valves (2-DRV-74-525A or 525C) were to be opened to drain the reactor cavity and to be closed to secure the draining evolution. Contrary to the above, on March 24, 2011, the operators failed to adequately implement 2-OI-74, Residual Heat Removal System, when they opened an additional RHR pump suction drain valve and failed to close that same valve upon system realignment on March 25. The mispositioned RHR Pump A Suction Drain Valve, 2-DRV-74-525A, resulted in an uncontrolled loss of RPV level during a routine suppression pool draindown evolution on March 25. However, because the finding was determined to be of very low safety significance and has been entered into the licensee's CAP as PER 344533, this violation is being treated as an NCV consistent with the Enforcement Policy. This NCV is identified as NCV 05000260/2011002-01, Loss of Reactor Water Level during Unit 2 Reactor Reassembly due to a Mispositioned Valve.

.2 Unit 3 Forced Shutdown Due To Automatic Scram

a. Inspection Scope

On December 26, 2010, Unit 3 entered an unplanned forced shutdown due to a manual reactor scram (see Section 4OA3.2). Operators commenced restart of Unit 3 (i.e., entered Mode 2) on January 6, and achieved full power on January 8. During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment and outage management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Plant Oversight Review Committee (PORC) event review and restart meeting on January 3 and 4, 2011.
- Closeout inspection of the Unit 3 Drywell and review of 3-GOI-200-2, Primary Containment Initial Entry and Closeout on January 4, 2011
- Reactor startup and power ascension activities per 3-GOI-100-1A, Unit Startup
- Reactor vessel and coolant heatup per 3-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage and attended management review committee meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions and/or reviewed completed test data for the following four 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.

In-Service Tests:

- 1-SR-3.5.3.3, RCIC Rated Flow at Normal Operating Pressure

Routine Surveillance Tests:

- 0-SR-3.6.4.1.3, Combined Zone Secondary Containment Drawdown and Integrity Test
- 3-SR-3.3.5.1.2(ATU B), Core and Containment Cooling Systems Analog Trip Unit Functional Test

Reactor Coolant System Leak Detection Tests:

- 2-SI-3.2.74(RHR II), Pressure Isolation Valve Leakage Test RHR Loop II Shutdown Cooling Return

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On February 2, 2011, the inspectors observed an Emergency Preparedness (EP) Severe Accident Management Guidelines (SAMG) drill that contributed to the licensee's Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicator (PI) measures to identify any weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room, 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 other applicable Emergency Plan Implementing Procedures. The inspectors also attended the licensee critique of the drill to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying weaknesses.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2RS1 Radiological Hazard Assessment and Exposure Control

a. Inspection Scope

Radiological Hazard Assessment - The inspectors reviewed a number of radiological surveys, including those performed for airborne areas, of locations throughout the facility including the Unit 2 drywell; Unit 1, Unit 2, and Unit 3 reactor buildings; the turbine building; and the independent spent fuel storage installation (ISFSI). The inspectors also walked down many of the same areas and selected radioactive material storage locations with a survey instrument, to evaluate material conditions, postings, and radiological controls. The inspectors observed jobs in radiologically risk-significant areas including high radiation areas and areas with, or without, the potential for airborne activity. The inspectors also evaluated surveys in relation to the identified hazards for sufficient detail and frequency

Instructions to Workers - During plant walk downs, the inspectors observed labeling and radiological controls on containers of radioactive material. The inspectors also reviewed radiation work permits (RWP) used for accessing high radiation areas and airborne areas, verifying that appropriate work control instructions and electronic dosimeter (ED) setpoints had been provided and to assess the communication of radiological control requirements to workers. The inspectors reviewed selected ED dose and dose rate alarms, to verify workers properly responded to the alarms and that the licensee's review of the events was appropriate. The inspectors observed pre-job RWP briefings and health physics technician coverage of workers. The inspectors reviewed the various methods being used to notify workers of changing or changed radiological conditions.

Contamination and Radioactive Material Control - The inspectors observed the release of potentially contaminated items from the radiologically controlled area (RCA) and from contaminated areas such as the drywell. The inspectors also reviewed the procedural requirements for, and equipment used to perform, the radiation surveys for release. During plant walk downs, the inspectors evaluated radioactive material storage areas and containers, including satellite RCAs and the low level radwaste facility, assessing material condition, posting/labeling, and control of materials/areas. In addition, the inspectors reviewed the sealed source inventory and verified labeling, storage conditions, and leak testing of selected sources. The inspectors verified if Category 1 and 2 sealed sources had been appropriately reported to the National Source Tracking System and physically verified the presence and controls of these sources.

Radiological Hazards Control and Work Coverage - The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress associated with the Unit 2 refueling outage. Established radiological controls were evaluated for selected tasks including CS and RHR pump motor replacement, torus desludging preparations, MSIV repair, and filter change out. The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results.

During walk downs with a radiation survey meter, the inspectors independently verified if ambient radiological conditions were consistent with licensee performed surveys, RWPs, and pre-job briefings; observed the adequacy of radiological controls; and observed controls for radioactive materials stored in the spent fuel pool. Inspectors also evaluated ED alarm set points and worker stay times against area radiation survey results for drywell and refueling floor activities.

Risk-Significant High Radiation Area and Very High Radiation Area Controls - The inspectors discussed the controls and procedures for locked-high radiation areas (LHRAs) and very high radiation areas (VHRAs) with health physics supervisors and the radiation protection manager. During plant walk downs, the inspectors verified the posting/locking of LHRA/VHRA areas.

Radiation Worker Performance and Radiation Protection Technician Proficiency - The inspectors observed radiation worker performance through direct observation, via remote camera monitoring, and via telemetry. Jobs observed associated with the outage included CS and RHR pump motor replacement, setup for diving for torus desludging, electrical disconnection of an MSIV, and a reactor water cleanup system filter changeout. These jobs were performed in high radiation, airborne, and/or contaminated areas. The inspectors also observed health physics technicians providing field coverage of jobs and providing remote coverage.

Problem Identification & Resolution – Licensee CAP documents associated with radiation monitoring and exposure control were reviewed and assessed. This included the review of selected PERs related to radworker and health physics technician performance. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NPG-SPP-3.1, Corrective Action Program, Rev. 1. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2RS01 of the Attachment.

Radiation protection activities were evaluated against the requirements of UFSAR Section 12; TS Sections 5.4 and 5.7; 10 Code of Federal Regulations (CFR) Parts 19 and 20; and approved licensee procedures. Radiological control activities for ISFSI areas were evaluated against 10 CFR Part 20, 10 CFR Part 72, and TS details. Records reviewed are listed in Section 2RS1 of the Attachment.

The inspectors completed one sample, as described in Inspection Procedure (IP) 71124.01.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls

a. Inspection Scope

Radiological Work Planning - Inspectors evaluated the As Low As Reasonably Achievable (ALARA) program guidance and implementation for ongoing tasks associated with U2R16. Inspectors also evaluated tasks and review of post-outage ALARA activities associated with U1C8 refueling outage. A list was obtained from the licensee of work activities for the current U2R16 outage. Inspectors selected five work activities to evaluate the ALARA Planning Report and associated documentation for jobs such as valve repair and maintenance, scaffold installation and removal, RHR motor maintenance, work on the refuel floor, and insulation and shielding installation. Inspectors also verified dose mitigation features, dose goals and other factors that went into planning the dose goal for each task. Associated RWPs were reviewed by inspectors to verify the integration of ALARA requirements into the documents for worker instruction. Inspectors followed the progression of the work activities to compare dose rates accrued and work evolution as compared to the ALARA planning. Since inspectors were not onsite through the end of the outage, post job reviews from earlier outages were reviewed and related PERs were noted.

Verification of Dose Estimates and Exposure Tracking Systems - Five ALARA work packages were reviewed by inspectors, including the assumptions and basis for the current collective exposure estimates. The ALARA procedures were also reviewed by inspectors. Inspectors spoke with ALARA personnel and reviewed daily logs that tracked and trended the dose of ongoing work. The ALARA trigger point work package reviews were also evaluated by inspectors.

Source Term Reduction and Radiation Worker – Performance - Inspectors reviewed licensee documents, records and spoke with HP personnel to review and evaluate source term reduction methods. Inspectors reviewed actions that had already been executed by the licensee such as elemental cobalt removal, valve replacements, water chemistry improvements. Inspectors also reviewed future plans such as shielding, turbine blade replacements and control rod replacements as part of the licensee's continued source term reduction efforts.

Radiation workers were observed both by remote monitoring and direct observation by inspectors. Workers observed doing JOG valve replacements, RHR motor pump replacements, and setting up for torus diving to perform desludging efforts, demonstrated ALARA philosophy.

Problem Identification and Resolution - Inspectors reviewed Licensee corrective action documents associated with ALARA planning and controls. This included review of selected PERs, self-assessments and audits. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NPG-SPP-3.1. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2RS2 of the Attachment.

Radiation worker performance was evaluated against the requirements of UFSAR Section 12; TS Sections 5.4 and 5.7; 10 CFR Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in Section 2RS2 of the Attachment.

The inspectors completed one sample, as described in IP 71124.02.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation

a. Inspection Scope

Engineering Controls - The inspectors reviewed the use of temporary and permanent engineering controls to mitigate airborne radioactivity during the U2R16 refueling outage. The inspectors observed the use of negative pressure units (NPUs) and containment enclosures during abrasive decontamination activities and breach of potentially contaminated valves. The inspectors also reviewed NPU testing certificates affixed to units in the reactor building and turbine building. The inspectors evaluated the effectiveness of continuous air monitors and air samplers placed in work area “breathing zones” to provide indication of increasing airborne levels.

Respiratory Protection Equipment - The inspectors reviewed the use of respiratory protection devices to limit the intake of radioactive material. This included review of devices used for routine tasks and devices stored for use in emergency situations. The inspectors reviewed ALARA evaluations for the use of respiratory protection devices during breach of potentially contaminated valves. Selected Self-Contained Breathing Apparatus (SCBA) units and negative pressure respirators (NPR)s staged for routine and emergency use in the Main Control Room and other locations were inspected for material condition, SCBA bottle air pressure, number of units, and number of spare masks and air bottles available. The inspectors reviewed maintenance records for selected SCBA units for the past two years and evaluated SCBA and NPR compliance with National Institute for Occupational Safety and Health certification requirements. The inspectors also reviewed records of air quality testing for supplied-air devices and SCBA bottles.

The inspectors observed the use of negative pressure respirators during contaminated valve work. The inspectors reviewed training curricula for various types of respiratory protection devices and interviewed radiation workers and control room operators on use of the devices including SCBA bottle change-out and use of corrective lens inserts. Respirator qualification records and medical fitness cards were reviewed for several Main Control Room operators and emergency responder personnel in the Maintenance and HP departments. In addition, qualifications for individuals responsible for testing and repairing SCBA vital components were evaluated through review of training records.

Problem Identification and Resolution – The PERs associated with airborne radioactivity mitigation and respiratory protection were reviewed and assessed. The inspectors also evaluated the licensee's ability to identify and resolve the issues in accordance with procedure NPG-SPP-03.1. The inspectors discussed the scope of the licensee's internal audit program and reviewed recent assessment results with the licensee.

Licensee activities associated with the use of engineering controls and respiratory protection equipment were reviewed against TS Section 5.4; 10 CFR Part 20; RG 8.15, Acceptable Programs for Respiratory Protection; and applicable licensee procedures. Documents reviewed during the inspection are listed in Section 2RS3 of the report Attachment.

The inspectors completed one sample, as described in IP 71124.03.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment

a. Inspection Scope

External Dosimetry - The inspectors reviewed National Voluntary Laboratory Accreditation Program (NVLAP) certification data (including TLD testing for neutron, gamma, and beta exposures) and discussed program guidance with licensee personnel for storage, processing, and results for active and passive personnel dosimeters currently in use. Licensee evaluations for shallow and deep dose assessments for workers with identified skin contaminations were reviewed and discussed. Comparisons between ED and personnel dosimeter data were discussed in detail with the licensee. In addition, the inspectors evaluated the use of extremity dosimetry, multi-badging, and re-positioning of whole body dosimetry during U2R16 maintenance activities.

Internal Dosimetry - Program guidance (including DAC-hr tracking), instrument detection capabilities, and assessment results for internally deposited radionuclides were reviewed in detail. The inspectors reviewed selected routine and investigative in vivo (Whole Body Count) analyses from January 2010 to January 2011. In addition, capabilities for collection and analysis of special bioassay samples were evaluated and discussed with licensee staff.

Special Dosimetric Situations - The inspectors evaluated the licensee's use of multi-badging, extremity dosimetry, and dosimeter relocation within non-uniform dose rate fields and discussed worker monitoring in neutron areas with licensee staff. The inspectors also reviewed records of monitoring for declared pregnant workers since January 2009 and discussed monitoring guidance with licensee staff. In addition, the adequacy of shallow dose assessments for selected Personnel Contamination Events occurring between January 2010 and January 2011 were reviewed and discussed with the licensee.

Problem Identification and Resolution - The inspectors reviewed and discussed selected CAP documents associated with occupational dose assessment. The inspectors evaluated the licensee's ability to identify and resolve the identified issues in accordance with procedure NPG-SPP-03.1. The inspectors also discussed the scope of the licensee's internal audit program and reviewed recent assessment results with the licensee.

Health Physics program occupational dose assessment activities were evaluated against the requirements of UFSAR Section 7; TS Section 5.4; 10 CFR Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in Section 2RS01, 2RS02, and 2RS04 of the report Attachment.

The inspectors completed one sample, as described in Inspection Procedure (IP) 71124.04.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Barrier Integrity

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 2010. 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.

- 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.

.2 Cornerstone: Occupational Radiation Safety

a. Inspection Scope

The inspectors reviewed PI data collected from January 1, 2010 through January 31, 2011, for the Occupational Exposure Control Effectiveness PI. For the reviewed period, the inspectors assessed CAP records to determine whether high radiation area, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred during the review period. In addition, the inspectors reviewed selected personnel contamination event data, internal dose assessment results, and ED alarms for cumulative doses and/or dose rates exceeding established set-points. The reviewed data were assessed against guidance contained in Nuclear Energy Institute 99-02. The reviewed documents relative to these PI reviews are listed in Sections 2RS1 and 4OA1 of the Attachment.

b. Findings

No findings were identified.

.3 Cornerstone: Public Radiation Safety (PS)

a. Inspection Scope

The inspectors reviewed the Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences PI results from January 1, 2010 through January 31, 2011. The inspectors reviewed PERs, effluent dose data, and licensee procedural guidance for classifying and reporting PI events. Reviewed documents are listed in Sections 4OA1 of the Attachment.

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 corrective action program (CAP). This review was accomplished by reviewing daily Service Request (SR) report summaries, and periodically attending Corrective Action Review Board (CARB) and PER Screening Committee (PSC) meetings.

.2 Annual Follow-up Sample - Operating Experience Smart Sample FY 2010-01 "Recent Inspection Experience for Components Installed Beyond Vendor Recommended Service Life."

a. Inspection Scope

The inspectors reviewed licensee components and processes in accordance with Operating Experience Smart Sample (OpESS) FY 2010-01 "Recent Inspection Experience for Components Installed Beyond Vendor Recommended Service Life." The inspectors verified the licensee's Preventive Maintenance (PM) programs addressed component aging, and reviewed the implementation of the program in accordance with licensee procedures; NPG-SPG-06.2, Preventive Maintenance and NPG-SPP-09.18, Integrated Equipment Reliability Program. Additionally, the inspectors interviewed the PM coordinator on the specifics of how components are managed, tracked and maintained with regards to service life. The inspectors focused on specific components and corrective actions associated with PER 305861, Fuel and Oil Filters In-service Beyond Manufacturers Recommendations; WO 08-719568-000, 2A Residual Heat Removal Pump Motor Upper Bearing Replacement; and PER 220336, Unit 3 Primary Containment Isolation System CR120A Relays not replaced during U3C14 Refueling Outage.

b. Findings

Introduction: An NRC identified Green non-cited violation of 10 CFR 50 Appendix B, Criteria XVI, Corrective Action, was identified for the licensee's failure to correct a condition adverse to quality related to Unit 3 primary containment isolation system (PCIS) logic relays exceeding their in-service life expectancy. Specifically, the licensee failed to replace numerous Unit 3 CR120A PCIS logic relays prior to exceeding the vendor recommended service lifetime.

Description: The licensee originally made an NRC commitment (Licensee Tracking Number NCO910037002 referencing LER 260/91001 for Unit 3) to replace all CR120A relays prior to Unit 3 restart in 1995 and to replace the relays at a specified frequency earlier than the vendor's recommended service life per GE SIL 229, "GE CR120A Relay Aging". The vendor recommended service life for the PCIS logic relay application was 15.83 years. All the Unit 3 PCIS relays were replaced prior to restart in 1995. Consequently, the licensee's preventive maintenance (PM) schedule required all of the relays to be replaced by July 1, 2010. Thirty four Unit 3 relays were originally scheduled to be replaced during the U3R14 refueling outage. However, 17 of the 34 relays were deleted from the outage scope without a proper evaluation. Of the 17 relays replaced, one was found degraded, which was indicative of relay aging. These PCIS relays were normally energized, and would typically be replaced during an outage because of the risk associated with replacing these relays while online which could result in an inadvertent reactor scram, PCIS group isolation, or engineered safeguards feature (ESF) actuation. To address this situation, the licensee initially planned to request the vendor to perform additional analysis and testing to extend the service life of the relays based on the actual environmental conditions, but this testing was disapproved. The licensee then determined that since the PCIS relays would fail in a safe condition, continued operation was justified beyond the expected service life. However, the inspectors expressed a concern that operating these relays until they failed could cause a significant plant upset or transient (e.g., scram, PCIS group isolation or accident signal actuation). Furthermore, these relays were classified as safety related and considered critical (CC1) by the licensee. According to licensee procedure NPG-SPP-9.18, Integrated Equipment Reliability Program, a run-to-failure methodology for safety related critical components was not acceptable.

Contributing to this PM issue were several work management problems. During this timeframe, PMs for these relays were deferred without technical justification and without adequate documentation. PMs were also not flagged in the work tracking system as mandatory or regulatory items. The licensee had initiated PER 220336 to address the relays that were deleted from the outage, which was flagged as an operability issue. However, PER 220336 was later closed out with no corrective actions because the licensee concluded that overdue relays were still operable since a relay failure (i.e., de-energization) would not prevent, but actually result in a safety function actuation. The licensee had multiple opportunities to properly disposition these relays with the CAP, and/or PM program, prior to exceeding the vendor recommended lifetime. In addition to the U3R14 refueling outage, the licensee missed two opportunities to replace the relays during unplanned forced outages in August and December of 2010. As of March 2011, the inspectors determined that 11 of the 17 relays had exceeded the vendor recommended service life, with the remaining six relays exceeding their service life by August 2011. This adverse condition to quality was not recognized by the licensee and as such not entered into their CAP. The licensee has now entered this issue into their CAP as PER 348160.

Analysis: The inspectors determined that the licensee's failure to replace or extend the service life of aging Unit 3 CR120A PCIS relays was a performance deficiency which resulted in 11 relays remaining in operation beyond their recommended service life. This finding was determined to be more than minor because the failure to replace the aged CR120A relays was associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the frequency of those events that upset plant stability and challenge critical safety functions during power operations. Failure of these relays could cause a scram, ESF actuation, and/or Group 1, 2, 3, or 6, primary containment isolation. The significance of the finding was evaluated using Phase 1 of the significance determination process (SDP) in accordance with the Inspection Manual Chapter (IMC) 0609 Attachment 4, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross cutting aspect of Appropriate Corrective Actions in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee failed to implement adequate corrective actions as part of PER 220336 to replace or extend the service life of the Unit 3 CR120A PCIS relays prior to exceeding their recommended service lifetime [P.1(d)].

Enforcement: 10 CFR 50, Appendix B, Criteria XVI, states, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the licensee failed to identify and initiate appropriate corrective actions to address the aging of Unit 3 CR120A PCIS relays before exceeding their recommended service life. This issue has been entered into the licensee's corrective action program as PER 348160. However, because the finding was determined to be of very low safety significance (Green) and has been entered into the licensee's CAP as PER 348160, this violation is being treated as an NCV consistent with the Enforcement Policy. This NCV is identified as NCV 05000296/2011002-02, Inadequate Corrective Actions To Address Unit 3 CR120A PCIS Relays That Exceeded Their Recommended Service Life.

c. Observations

No findings were identified with the Diesel Generator Lube Oil Filters or the 2A residual heat removal pump maintenance. However, the inspectors had the following observations concerning the EDG's:

- Inspectors reviewed the licensee's functional evaluation for the lube oil filters regarding filter operability beyond the vendor recommended service life for replacement. The inspectors identified that the functional evaluation did not address the primary failure mechanism as described in the vendor manual (VTD-E147-0200), in which the center tube collapses resulting in the engine oil bypassing the filters. Also, written instructions were not provided to the licensee's central laboratories requesting specific inspections of the lube oil filters, including anticipated degradation. Although, the licensee's functional evaluation was subsequently revised, it did not result in a change in the overall conclusion.

- The EDG functional evaluation required periodic oil sampling to detect oil filter degradation prior to each monthly run beyond the 30 month manufacturer recommended lifetime. The inspectors determined that these samples were not being performed. The licensee initiated PER 332999. The licensee also determined that the failure to take lube oil samples was a work management issue.

- Mission times documented in EDG functional evaluations conflicted with the mission time documented in Engineering Work Request (EWR No. 09-MEB-999-023). The EWR defined mission times for various safety related SSCs. The functional evaluation mission time was 24 hours. The EWR mission time for EDGs was 30 days. The licensee initiated PER 328682 to address this conflicting information.

.3 Annual Follow-up Sample – Cross Cutting Aspect H.4(b) Theme

a. Inspection Scope

The inspectors reviewed the cause analysis and specific corrective actions associated with PER 292396, Evaluate Recent HU Cross Cutting Violations Relative to PER 228347, Emerging Trend in Human Performance Cross-Cutting Area. This PER was initiated to evaluate recent NCVs with human performance cross cutting aspects (CCA) identified. The original focus of PER 228347 was to address a previously identified H.2.c CCA trend that was reviewed by the inspectors as documented in IR 05000259, 260, 296/2010005. Since then five additional NCVs with CCAs in the area of Human Performance were identified. The licensee initiated PER 292396 to evaluate these NCVs and develop any further corrective actions that may be deemed necessary.

As defined by Inspection Manual Chapter (IMC) 0310, two of these five NCV's were related to the CCA of Adequate and Complete Documentation [H.2.c] in the Resources component of the Human Performance area, as follows: (1) Failure to establish an adequate procedure during testing of pressure switches on the HPCI system (PER 239313); and (2) Inadequate preventive maintenance procedures for Siemens vacuum circuit breakers (PER 234443). The three other NCV's of PER 292396 were related to the CCA of Procedural Compliance [H.4.b] in the Work Practices component of the Human Performance area, as follows: (1) Improper execution of the waiver process of the fatigue rule (PER's 225197 and 225391); (2) Inadequate risk assessment of online risk with ongoing maintenance (PER's 241885 and 254000); and (3) Failure to implement applicable provisions of preventive maintenance program (PER 247960).

The inspectors also reviewed the five specific NCVs, including associated PERs, and the root cause analysis (RCA) and corrective actions of PER 228347 to verify the licensee's conclusion in PER 292396, that these NCV's were encompassed by the corrective actions of PER 228347.

b. Observations and Findings

No findings were identified. In general, the inspectors determined that the licensee's evaluation of the NCV's associated with PER 292396, was thorough and consistent with the licensee's processes. The licensee's evaluation of the five additional NCV's was determined to have adequately demonstrated these findings were bounded by the common cause RCA of PER 228347. Inspectors also concluded the RCA associated with PER 228347 adequately addressed operability, reportability, common cause, generic concerns, extent-of-condition, and extent-of-cause for the associated Human Performance events for the additional NCVs. Furthermore, the inspectors also determined that the licensee had appropriately identified and prioritized corrective actions to encompass these Human Performance issues. In general, the corrective actions to prevent recurrence (CAPRs) and additional corrective actions implemented to date, or scheduled to be implemented, were considered reasonable to address the human performance related cause(s). However, the licensee's corrective actions for the adverse trend in human performance were still ongoing, a final effectiveness review of these actions has not been accomplished.

.4 Annual Follow-up Sample - Corrective Actions Associated with RHRSW/EECW Pump In Service Testing per ASME Code

a. Inspection Scope

The inspectors reviewed the specific corrective actions for licensee initiated PER 225844 associated with NCV 05000259, 260, 296/2010002-05, Untimely Corrective Actions to Restore Compliance of EECW Pump In-Service Testing with ASME Operations and Maintenance of Nuclear Power Plants (OM) Code Requirements,.

The inspectors reviewed the licensee's corrective action plans and interviewed engineering personnel to assess the effectiveness and adequacy of the licensee's efforts to correct the inspector identified problems regarding licensee conformance with ASME OM Code requirements. The inspectors focused their review on the effectiveness of the licensee's corrective actions taken to address the conditions identified, including subsequent operability evaluations; the extent of condition analysis; and the prioritization of the corrective actions. Additionally, the inspectors evaluated these elements against regulatory requirements and the licensee's CAP.

b. Assessment and Observations

No findings were identified. However, the inspectors made the following observation:

In April 2010, the inspectors identified NCV 2010002-05 regarding deficiencies associated with instrument inaccuracies, baseline development methodology, and inability to trend pump degradation. The PER 225844 corrective actions resulted in the procurement of new digital test instrumentation meeting the accuracy of the ASME code, elimination of multiple reference values for summer and winter conditions, development of single reference values, and revised testing procedures. Since the implementation of the new procedures, instrumentation, and acceptance criteria in October, 2010, there have been no EECW/RHRSW pump failures in the subsequent four months. At no time did any of the prior EECW/RHRSW pump acceptance criteria failures result in less than the minimum required flows for design basis accident conditions.

The licensee also determined that seasonal variations of EECW and RHRSW pump performance occurred due to changes in river water temperature. This caused changes in internal pump clearances (impeller gap) due to differences in the thermal expansion coefficients of the stainless steel pump shaft and the carbon steel pump casing. Inspectors noted that the root cause of the pump performance variations was not corrected and that the licensee expected there will be future acceptance criteria failures during seasonal temperature changes. The licensee determined that modification of existing EECW and RHRSW pumps with closed vane pumps would mitigate seasonal variations in pump performance. Corrective actions associated with these modifications were identified under PER 156818 as enhancements. The licensee's CAP allowed PERs to be closed if the only outstanding actions were classified as enhancements, therefore, PER 156818 was closed. However, the inspectors questioned the licensee regarding how the planned modifications were being tracked. As a result of this inspector observation, the licensee initiated PER 357431 to document corrective actions to modify the open vane RHRSW/EECW pumps to closed vane.

.5 Annual Follow-up Sample – TS 5.5.2, Primary Coolant Sources Outside Containment

a. Inspection Scope

The inspectors reviewed the licensee's program for TS 5.5.2, Primary Coolant Sources Outside Containment, and its implementation. The inspectors also reviewed licensee's programs for TS 5.5.5, Component Cyclic or Transient Limit, and TS 5.5.13, Control Room Envelope Habitability Program.

b. Observations and Findings

One finding (see below) and one observation were identified by the inspectors. Inspectors observed that the number of allowed shutdown cycles listed in 0-TI-19, Reactor Vessel Fatigue Usage Factor Evaluation, Appendix B, Table 1, Summary of Cycles Covered By Stress Report, indicated 120, whereas UFSAR 4.2.5, Safety Evaluation, identified the number of shutdown cycles as 118. The licensee initiated PER 348278 to resolve the difference.

Introduction: An NRC identified Green NCV of TS 5.5.2, Primary Coolant Sources Outside Containment was identified for the licensee's failure to establish, implement and maintain an adequate program for minimizing primary coolant leaks from systems (i.e., CS, RHR, HPCI, and RCIC) outside containment, that could contain highly radioactive fluids during a serious transient or accident, to levels as low as practicable.

Description: On December 20, 2010, the inspectors identified a leak of approximately 70 drops per minute leak from the body to bonnet flange of the 2C RHR pump minimum flow valve (2-SHV-074-0087). The licensee initiated PER 300003 to address the inspector's concerns regarding compliance with TS 5.5.2. After further review of other known CS, RHR, HPCI, and RCIC system leaks the inspectors concluded the licensee was not systematically evaluating and repairing these type of leaks consistent with TS 5.5.2. Also, the inspectors determined that existing procedures, including NPG-SPP-06.8, Leak Reduction Program, did not establish the periodic visual inspection or system leak test requirements specifically prescribed by TS 5.5.2.a. and b., respectively. Although, SPP-06.08, did classify the CS, RHR, HPCI, and RCIC systems as Category 1 systems and establish a prioritization scheme for Maintenance Action Levels (MALs) to repair identified leaks. The licensee's execution of the SPP-6.8 MALs was inconsistent in that numerous leaks (small, medium and large) from these Category 1 systems were not consistently repaired "before the next work-week cycle." Furthermore, the licensee was not evaluating the aggregate volume of these leaks to ensure the assumptions used in their design basis accident (DBA) safety analyses for primary coolant leakage into secondary containment was being maintained. According to the UFSAR Section 14.6.3.5, Fission Product Release From Secondary Containment, the licensee assumed 20 gpm primary coolant leakage into secondary containment.

The licensee's immediate corrective actions included prompt identification, evaluation, and prioritization of all known primary coolant leaks outside containment from the CS, RHR, HPCI, and RCIC systems. The aggregate of these leaks were subsequently determined to be approximately 1.24, 0.96, and 0.16 gpm for Units 1, 2, and 3, respectively, which were considerably less than the UFSAR assumed leakage. In addition, the licensee developed a new program 0-TI-578, Minimizing Primary Coolant Sources Outside Containment.

Analysis: The inspectors determined that the licensee's failure to establish, implement, and maintain an adequate program to meet the requirements of TS 5.5.2 constituted a performance deficiency. This performance deficiency was determined to be more than minor because if left uncorrected it could have the potential of leading to a more significant safety concern. Specifically, the licensee's failure to effectively minimize and monitor primary coolant leakage outside containment could have resulted in increased main control room exposure and/or offsite dose during an accident due to excessive radioactive fission product releases into secondary containment. The finding was determined to be of very low safety significance (Green) according to IMC 0609, Appendix H, Containment Integrity Significance Determination Process, Section 6.0, Type B Findings, because the primary coolant leak rate into secondary containment was a small fraction of the leakage assumed in the DBA safety analyses.

The cause of this finding was directly related to the cross-cutting aspect Complete and Accurate Procedures in the Resources component of the Human Performance area because the licensee's existing procedures were inadequate and incomplete for addressing the program requirements of TS 5.5.2 (H.2.(c)).

Enforcement: Technical Specifications (TS) 5.5.2. Primary Coolant Sources Outside Containment, in part, required the licensee to establish, implement and maintain a program for minimizing primary coolant leaks from systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to levels as low as practicable. This program was required to include periodic visual inspection and system leak test requirements for each system. Contrary to the above, the licensee failed to establish a program for minimizing primary coolant leaks from systems outside containment that required periodic visual inspection and system leak test requirements for each system since the implementation of TS Amendments 234 for Unit 1, 253 for Unit 2, and 212 for Unit 3, on July 15, 1995. However, because the finding was determined to be of very low safety significance (GREEN) and has been entered into the licensee's CAP as PER 317464, this violation is being treated as an NCV consistent with the Enforcement Policy. This NCV is identified as NCV 05000259, 260, and 296/2011002-03, Inadequate TS 5.5.2. Program for Primary Coolant Leaks Outside Containment.

.6 Annual Follow-up Sample – Nuclear Fatigue Rule

a. Inspection Scope

The inspectors reviewed the results of Quality Assurance (QA) audit SSA 1010, Fatigue Rule Site Audit Report, for the Browns Ferry Nuclear Plant dated October 7, 2010; and QA audit SSA 1010, Fatigue Rule Fleet Comparative Audit Report, dated October 25,

2010. The inspectors reviewed the applicable PER 255792, and associated RCA, including related background information. The inspectors interviewed the audit team leader and RCA team leader. In addition, the inspectors attended and reviewed the followup QA audit SSA1109, Fatigue Rule Audit Exit Debrief and met with the site Nuclear Fatigue Rule (NFR) subject matter expert (SME) to discuss corrective action plans to address recurring NFR violations.

b. Findings

One finding was identified. The enforcement aspects of this finding are discussed in Section 4OA7.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000259/2010-002-00, Drywell Pressure Instrument Channel Inoperability Due to Improper Instrument Tubing Slope

a. Inspection Scope

The inspectors reviewed the LER and associated PERs, including applicable apparent cause determinations and functional evaluations.

On September 16, 2010, the licensee determined that a condition initially identified on December 8, 2008, resulted in the inoperability of a drywell pressure channel instrument for longer than allowed by TS. This LER actually described two events in which two separate drywell pressure instruments were inoperable longer than allowed by the TS. The first event was associated with instrument 1-PT-064-56C. The licensee determined that this instrument was inoperable from October 2, 2008, to December 9, 2008. However, during this time Unit 1 was shutdown between October 25 and November 30, 2008. For the period of applicable channel inoperability, the licensee did not meet TS Limiting Condition of Operation (LCO) 3.3.1.1, Reactor Protection System Instrumentation; 3.3.6.1, Primary Containment Isolation System Instrumentation; 3.3.6.2 Secondary Containment Isolation System Instrumentation; and 3.3.7.1 Control Room Emergency Ventilation System Instrumentation. The second event was associated with instrument 1-PIS-064-56B. The licensee determined that this instrument was inoperable from May 25, 2010 to October 6, 2010. During this channel inoperability, the licensee failed to meet TS LCOs 3.3.1.1; 3.3.6.1; 3.3.6.2; 3.3.7.1; 3.3.3.1, Post Accident Monitoring Instrumentation and 3.3.3.2 Backup Control System.

For both events, the licensee determined the direct cause was due to an improper instrument line slope that allowed water to accumulate in the sensing line which resulted in a non-conservative biased instrument output. The root cause of the improperly sloped instrument sensing lines was the licensee’s failure to adhere to engineering specification N1E-003, Instrument and Instrument Line Installation and Inspection. Corrective actions included a blowdown of the affected instrument line to remove the entrained water. Also, a design change was implemented during the most recent refueling outage which re-routed the affected instrument sensing line to eliminate the water traps in accordance with engineering specification N1E-003.

b. Findings

Two findings were identified. The enforcement aspects of these findings are discussed in Section 4OA7. This LER is considered closed.

.2 (Closed) Licensee Event Report (LER) 05000296/2010-004-00, Manual Reactor Scram Due to High Vibration on the Generator Exciter Inboard and Outboard Journal Bearings

a. Inspection Scope

On December 26, 2010, Unit 3 was manually scrammed from approximately 90 percent RTP due to high vibrations on the main generator exciter bearings that exceeded the required threshold for tripping the main turbine. The initial follow-up of this event by the inspectors was documented in Section 1R20.2 of IR 05000296/2010005. The inspectors reviewed the applicable LER that was issued on February 24, 2011, and it's associated PER 301505, which included the cause determination and corrective actions. The licensee concluded that the direct cause of the high vibrations on the exciter bearings was due to reduced clearances at the exciter bearing oil and air deflectors caused by an excessive temperature differential between the two air coolers.

b. Findings

This LER is considered closed with one finding identified.

Introduction: A Green self-revealing finding (FIN) was identified for the licensee's failure to adequately evaluate and take the required actions established by site standards to address an adverse system performance trend that had degraded below acceptable levels associated with the main generator exciter air coolers. This was a direct contributor to the Unit 3 manual scram on December 26, 2010, due to high vibrations on the main generator exciter bearings.

Description: On December 26, 2010, operators initiated a recirculation pump runback, and then a manual reactor scram, when high main turbine vibrations were indicated on the inboard and outboard main generator exciter journal bearings. This scram also tripped the Unit 3 main turbine. The inspectors reviewed the licensee's RCA report contained in PER 301505. The licensee determined the direct cause of the high vibrations on the exciter bearings was due to reduced clearances at the exciter bearing oil and air deflectors caused by an excessive temperature differential between the two air coolers. The licensee concluded that the root cause of the high vibrations was due to inadequate operational guidance of the main generator exciter air cooling system. The licensee also identified a contributing cause regarding inadequate system monitoring in that the temperature differential of the main generator exciter air coolers had exceeded the licensee-defined limit of 10F in September 2010 was identified by the licensee but no PER was initiated to address the condition. Licensee procedure NEDP-20, Conduct of the Engineering Organization, Rev .12, Section 3.1, System Performance Monitoring, required initiation of a PER when system performance has degraded below acceptable levels or an adverse trend was detected. The licensee developed corrective actions that installed vents on the exciter air coolers to minimize air binding, established a process and frequency for venting the exciter air coolers, and increased the oversight of the system monitoring process within the Engineering organization by adding a supervisory review to the system monitoring process.

Analysis: The inspectors determined that the licensee's failure to adequately evaluate and take required actions to address an adverse system performance trend that had degraded below acceptable levels for the main generator exciter air coolers was a performance deficiency. This finding was considered greater than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability. Specifically, the consequence of the finding directly contributed to a reactor scram on Unit 3. The significance of the finding was evaluated using Phase 1 of the SDP in accordance with the IMC 0609 Attachment 4, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting aspect of Corrective Action Program Implementation in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee failed to identify the adverse trend of excessive differential temperatures between the exciter air coolers in a timely manner and enter it into the corrective action program. (P.1.a).

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements since the main generator exciter air coolers and the portion of the raw cooling water system that supplies these coolers were not safety-related. However, because this finding does not involve a violation of regulatory requirements, was entered in the licensee's CAP as PER 301505, and has very low safety significance, it is identified as FIN 05000296/2011002-04, Failure to Identify Adverse Trend Resulted in Reactor Scram.

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 Independent Spent Fuel Storage Installation Walkdown

a. Inspection Scope

Under the guidance of IP 60855.1, as directed by IMC 2515, Appendix D, the inspectors reviewed the licensee's procedures and documentation regarding independent spent fuel storage installation (ISFSI) related activities to verify they met the commitments and requirements specified in the HI-STORM 100 Final Safety Analysis Report (FSAR); Certificate of Compliance (CoC) No.1014, including Appendix A, Technical Specifications, and Appendix B, Approved Contents and Design Features; and 10 CFR Part 72.210 for a general licensed ISFSI. In addition, the inspectors interviewed responsible personnel and witnessed selected ISFSI related activities, such as dry cask storage operations, to ensure that the licensee performed these activities in a safe and compliant manner consistent with approved procedures.

b. Findings

Introduction: A Severity Level IV, cited violation (VIO) of 10 CFR 72.212, Conditions of general license issued under §72.210, was identified by the inspectors for the licensee's repetitive failure to adequately control transient combustible materials stored in the proximity of loaded dry casks on the ISFSI pad in accordance with site procedures.

Description: On February 3, 2011, while performing a routine walkdown of the ISFSI enclosed area, the inspectors observed seven storage cradles, multiple storage pallets and storage devices or cribbing located on or near the dry cask storage pad. The cradles, pallets and cribbing, were all constructed of wood products. The nearest items, were wood cradles located approximately 10 to 15 feet from the closest HI-STORM cask loaded with spent fuel. The other wood storage devices were approximately 20 feet from the closest loaded cask and were located both on and off the ISFSI pad. No apparent work was in progress at the time of discovery. The inspectors contacted responsible licensee personnel who promptly removed all the transient combustible material from the ISFSI exclusion area and initiated PER 318694. The licensee also performed an evaluation of the transient combustible loading for this material.

This was the third occurrence identified by the inspectors of transient combustibles located in close proximity to HI-Storm casks loaded with spent fuel. The first two occurrences were on May 25, 2010 (see NCV 07200052/2010002-001, Transient Combustibles Stored Near Independent Spent Fuel Storage Facility in Excess of Amount Allowed), and on August 17, 2010 (see NOV 07200052/2010003-001, Transient Combustibles Stored Near Independent Spent Fuel Storage Facility in Excess of Amount Allowed), in both instances diesel fuel contained in vehicles left parked in close proximity to loaded HI-Storm casks was greater than the maximum allowed.

According to NPG-SPP-18.4.7, Control of Transient Combustibles, the requirements and controls for handling and use of transient combustibles in proximity of the BFN ISFSI/Dry Cask Storage Pad were contained within drawings 0-47E201-1 and 0-47E201-2. In particular, drawing 0-47E201-2, ISFSI Fire Hazards Analysis Compensatory Actions, Item 11 stated that wooden structures facing the ISFSI were limited to a front face maximum height of 15 feet and a maximum width of 24 feet for a surface area total of 360 square feet, at a distance of 30 feet from the edge of the closest HI-STORM. Furthermore, General Operating Instruction (GOI) 0-GOI-300-1/ATT-12, Outside Operator Round Log, required operators to perform an inspection daily to ensure the ISFSI Pad and exclusion area were clear of the following: Flammable material such as wood, rags and plastic sheeting. If the ISFSI pad and exclusion area were not clear of these materials, then report the results to the Unit 3 Supervisor for evaluation of acceptability in accordance with drawing 0-47E201-2. Per 0-GOI-300-1/ATT-12 the ISFSI Pad exclusion area is defined as within 150 feet of the edges of the ISFSI Pad in all directions.

Based upon discussion with the licensee and a review of work performed in the area, the inspectors determined that the licensee had allowed the wood cradles and cribbing to be left near a loaded HI-STORM cask for approximately one week from on or about January 26 to February 3, 2011. The licensee was performing work in the area to upright and inspect Multi Purpose Containers for the upcoming campaign. However, plant operators had not notified the Unit 3 US of the stored wooden material, and no evaluation had been performed on the acceptability of the transient combustible material as required by 0-GOI-300-1/ATT-12. Subsequent calculations by the licensee determined that the radiative heat load of the wood items was only about five percent of the allowed transient combustible loading limit.

Analysis: The Reactor Oversight Process (ROP) was not used for this issue because inspections of ISFSI activities that do not involve the operating reactor plant are not addressed by the reactor safety cornerstones in the ROP's SDP. Therefore, this issue was evaluated using traditional enforcement as described in the NRC Enforcement Policy. This issue was greater than minor because it was associated with the protection against potential fire damage to the stored spent fuel which if left uncorrected, could become a more significant safety concern. The prolonged presence of combustible materials in the vicinity of the stored spent fuel could increase the vulnerability of the casks to a fire and therefore increase the potential likelihood of fuel damage and/or release during a fire event. This finding was not considered to be a significantly appreciable threat for potential exposures to, or release of, radiation because of the limited quantity of combustibles and the short durations of time they were stored unattended in the vicinity of the loaded dry casks. Therefore, the finding was determined to be of Level IV significance based on Example 6.2.d.2 of the NRC Enforcement Policy. No cross cutting aspect was assigned because the ROP was not applicable.

Enforcement: 10 CFR 72.212, Conditions of general license issued under §72.210, section (b)(9) stated, in part, that the licensee shall "Conduct activities related to storage of spent fuel under this general license only in accordance with written procedures." Procedure NPG-SPP-18.4.7, Control of Transient Combustibles, stated that requirements and controls for handling and use of transient combustibles associated with the BFN ISFSI/Dry Cask Storage Pad were contained within drawings 0-47E201-1 and 0-47E201-2. These drawings established limits for the amount of transient combustibles that could be stored in proximity to a loaded HI-STORM cask. In particular, drawing 0-47E201-2, ISFSI Fire Hazards Analysis Compensatory Actions, Item 11, stated that wooden structures facing the ISFSI were limited to a front face maximum height of 15 feet and a maximum width of 24 feet for a surface area total of 360 square feet, at a minimum distance of 30 feet from the edge of the closest HI-STORM. Furthermore, 0-GOI-300-1/ATT-12, Outside Operator Round Log, required operators to perform a daily inspection to ensure the ISFSI Pad and exclusion area were clear of flammable material such as wood. If the ISFSI Pad and exclusion area were not clear of flammable material (e.g., wood) then operators were to report to the Unit 3 supervisor for an evaluation in accordance with drawing 0-47E201-2.

Contrary to the above, on February 3, 2011, for approximately one week, seven storage cradles, storage pallets and multiple storage devices or cribbing all constructed from wood were discovered located on or near the dry cask storage pad within approximately 10 to 20 feet of the nearest loaded HI-STORM. The outside operator also failed to report the transient combustible material to the Unit 3 supervisor for evaluation as required by 0-GOI-300-1/ATT-12, and no transient combustible evaluation according to drawing 0-47E201-2 was performed. The licensee promptly removed the wood material from the ISFSI Pad exclusion area. This violation was determined to be a Severity Level IV violation and was entered into the licensee's corrective action program as PER 318694. This is a violation of 10 CFR 72.212 and is identified as VIO 07200052/2011002-05, Repeated Failure to Control Transient Combustibles in Proximity of the Independent Spent Fuel Storage Facility. A notice of violation is attached.

.3 Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)

a. Scope

The inspectors performed the TI concurrent with IP 71124.01 Radiation Hazard Analysis. The inspectors reviewed the licensee's source inventory records and identified the sources that met the criteria for reporting to the NSTS. The inspectors visually identified the sources contained in various calibration systems and verified the presence of the source by direct radiation measurement using a calibrated portable radiation detection survey instrument. The inspectors reviewed the physical condition of the irradiation devices to include documented source leak checks as appropriate. The inspectors verified that the plant startup sources were below reportable activity levels. Furthermore, the inspectors reviewed the licensee's procedures for source receipt, maintenance, transfer, reporting and disposal. The inspectors also reviewed documentation that was used to report the sources to the NSTS. Documents reviewed are listed in sections 2RS1 of the Attachment.

b. Findings

No findings were identified

This completes the Region II inspection requirements for TI 2515/179 at this site.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On January 11, 2011, the senior resident inspector presented the inspection results to Mr. Keith Polson, Site Vice President, 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. A re-exit was also conducted with Mr. Steve Bono and other members of the site's staff on May 13, 2011.

.2 OTHER ACTIVITIES

An interim exit with licensee management and staff was conducted on February 3, 2011, to discuss the results of the Modifications Inspection. Proprietary information reviewed by the team as part of routine inspection activities was returned to the licensee in accordance with prescribed controls.

On February 10, 2011, the inspectors presented the heat sink inspection results to the plant management and staff. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

On March 18, 2011, the inspectors discussed the results of the RP inspection with Mr. Keith Polson, – Browns Ferry Nuclear Plant, and other responsible staff.

4OA7 Licensee-Identified Violations

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

- Unit 3 Technical Specification 5.4.1.a. required that written procedures recommended in RG 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained. Procedures for performing maintenance on safety related equipment were specifically listed as recommended procedures by Section 9 of Regulatory Guide 1.33, Appendix A. Preventive maintenance procedure EPI-0-000-BKR15, 4KV Wyle/Siemens Horizontal Vacuum Circuit Breaker (Type-3AF) and Compartment Maintenance, was established in part to specifically inspect the rubber bumper located on the MJ (52STA) switch actuator for damage/indention that would reduce the amount of travel of the MJ switch when the breaker was actuated. Contrary to this, the licensee determined that preventive maintenance was not performed on 3ED 4KV Shutdown Board Normal Feeder Breaker 1342 within the required six year periodicity, and was deferred an additional two years without adequate technical justification. On February 10, 2011, while performing thermography on the 3ED 4KV Shutdown Board, the licensee identified several deenergized relays resulting from unmade contacts due to age-related wear on the MJ switch actuation arm rubber bumper which would have prevented an automatic start of 3D Residual Heat Removal (RHR), 3D Core Spray (CS), and D1 RHR

Service Water (RHRSW) Pumps. This finding was entered into the licensee's CAP as PER 324038 and PER 322640 for the associated cause analysis. The finding was determined to be of very low safety significance because it did not constitute a total loss of system or train safety function since the pumps would have automatically started with the diesel generator supplying the electric board and manually started regardless of the electrical source. Therefore, Train D RHR and CS safety functions would have been available during a design accident condition.

- Unit 1 Technical Specification 3.3.1.1, Reactor Protection System Instrumentation, required two operable drywell (DW) pressure channels per trip system in Modes 1 and 2, or place the inoperable channel in trip within 12 hours, or be in Mode 3 in 12 hours. Contrary to this, as described in the LER 259/2010-002, the licensee determined that DW pressure instrument 1-PT-064-56C was inoperable from October 2 until October 25, 2008, and again from November 30, 2008 until the December 9, 2008, due to accumulated water in the sensing line from improper line slope, without placing the channel in trip or shutting down Unit 1. This violation is not greater than Green because there was no loss of function due to the availability of redundant instrumentation. This issue was entered in the licensee's CAP as PERs 159710, 219150, 242068 and 279760.

- Unit 1 Technical Specification 3.3.1.1, Reactor Protection System Instrumentation, required two operable drywell (DW) pressure channels per trip system in Modes 1 and 2, or place the inoperable channel in trip within 12 hours, or be in Mode 3 in 12 hours. Contrary to this, as described in the LER 259/2010-002, the licensee determined that DW pressure instrument 1-PIS-064-56B was inoperable from May 25, 2010 to October 6, 2010, due to accumulated water in the sensing line from improper line slope, without placing the channel in trip or shutting down Unit 1. This violation is not greater than Green because there was no loss of function due to availability of redundant instrumentation. This issue was entered in the licensee's CAP as PERs 159710, 219150, 242068 and 279760.

- 10 CFR 26.205(d) states, in part, that the licensee shall control the work hours of personnel performing maintenance, as identified by 10CFR26.4(a)(4), to ensure maintenance personnel individually meet the work hour limits, minimum break periods, and minimum days off prescribed by 10CFR26.205(d)(1) thru (3). However, from April to November 2010, due a programmatic breakdown in tracking overtime hours, the licensee failed to control individual work hours, breaks, and minimum days off for numerous Maintenance Department personnel in a manner that complied with the aforementioned regulatory requirements and the licensee's program SPP-1.5 (later changed to NPG-SPP-03.21), Fatigue Management and Work Hour Limits. These violations were captured in the licensee's CAP as PERs 255792 and 322569. This finding was determined to be of very low safety significance (Green) because no significant human errors occurred that were attributable to fatigue while these individuals were in violation of the Nuclear Fatigue Rule.

SUPPLEMENTAL INFORMATION

KEY LICENSEE POINTS OF CONTACT

T. Albright, Simulator Manager
W. Baker, Operations Support Superintendent
S. Bono, Maintenance Manager
J. Boyer, System Engineering Manager
O. Brooks, Operations LOR Supervisor
W. Byrne, Site Security Manager
P. Chase, Site Nuclear Assurance Manager
T. Cole, Radiation Protection Operations ManagerJ. Colvin, Engineering Programs Manager
S. Cowan, Radiation Protection Technical Support Superintendent
P. Donahue, Assistant Engineering Director
G. Doyle, Assistant to the Site Vice President
M. Durr, Director of Engineering
M. Ellet, Maintenance Rule Coordinator
L. Ellgass, Cooperative Program Manager, MR/PM
J. Emens, Licensing Manager
S. Emmons Human Performance Manager
B. Evans, Instrumentation and Controls Superintendent
A. Feltman, Emergency Preparedness Manager
J. Ferguson, Radiation Protection Support Superintendent
N. Gannon, Plant General Manager
K. Gregory, Director Projects
K. Groom, Mechanical Design Engineering Supervisor
D. Hughes, Operations Manager
B. Jones, Mechanical Maintenance Superintendent
M. Keck, Reactor Engineering Manager
S. Kelly, Assistant Work Control Manager
R. King, Design Engineering Manager
D. Malinowski, Operations Training Manager
T. Marlow, Director of Safety and Licensing
M. McAndrew, Assistant Operations Manager
G. McConnell, Equipment Reliability Manager
J. Morris, Director Training
R. Norris, Radiation Protection Manager
W. Nurnberger, Work Control Manager
W. Pearce, Performance Improvement Manager
K. Polson, Site Vice President
M. Rasmussen, Operations Superintendent
H. Smith Fire Protection Supervisor
T. Smith, Component Engineering Manager
J. Underwood, Chemistry Manager
S. Walton, Electrical Maintenance Superintendent

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

07200052/2011002-05	VIO	Repeated Failure to Control Transient Combustibles in Proximity of the Independent Spent Fuel Storage Facility (Section 4OA5.2)

Opened and Closed

05000260/2011002-01	NCV	Loss of Reactor Water Level during Unit 2 Reactor Reassembly due to a Mispositioned Valve (Section 1R20.1)
05000296/2011002-02	NCV	Inadequate Corrective Actions To Address Unit 3 CR120A PCIS Relays That Exceeded Their Recommended Service Life (Section 4OA2.2)
05000259, 260, 296/2011002-03	NCV	Inadequate TS 5.5.2 Program for Primary Coolant Leaks Outside Containment (Section 4OA2.5)
05000296/2011002-04	FIN	Failure to Identify Adverse Trend Resulted in Reactor Scram (Section 4OA3.2)

Closed

05000259/2010-002	LER	Drywell Pressure Instrument Channel Inoperability Due to Improper Instrument Tubing Slope (Section 4OA3.1)
05000296/2010-004	LER	Manual Reactor Scram Due to High Vibration on the Generator Exciter Inboard and Outboard Journal Bearings (Section 4OA3.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

0-GOI-200-1, Freeze Protection Inspection, and associated checklists, Rev. 66, Dated 1/12/11 to 1/14/11
Narrative Logs, 1-12-2011, to 1-14-2011
PER 277476
PER 294448
PER 311771
SR 292873
SR 292885
SR 311104
SR 311105
WO 111722357
WO 111844682

Section 1R04: Equipment Alignment

2-SR-3.1.7.3 "SLC System Enriched Sodium Pentaborate Solution Concentration, Quantity Calculation & ATWS Equivalency Calculation"
NPG-SPP-06.9.1 "Conduct of Testing"
IP 71111.04 "Equipment Alignment"
3-TI-18 "Enriched Sodium Pentaborate (SPB) Solution Preparation Procedure for the Standby Liquid Control (SLC) System"
2-OI-063 "Standby Liquid Control System"
2-OI-063 Att. 1 "Standby Liquid Control (SLC) System Valve Lineup Checklist"
2-OI-063 Att. 2 "Standby Liquid Control (SLC) System Panel Lineup Checklist"
2-OI-063 Att. 3 "Standby Liquid Control (SLC) System Electrical Lineup Checklist"
2-OI-063 Att. 4 "Standby Liquid Control (SLC) System Instrument Inspection Checklist"
47E854 – Series "Flow Diagram for the Standby Liquid Control System
1-OI-75, Core Spray System, Rev. 23
1-OI-75/ATT-1, Attachment 1 Core Spray System Valve Lineup Checklist, Rev. 20
1-OI-75/ATT-2, Attachment 2 Core Spray System Panel Lineup Checklist, Rev. 20
1-OI-75/ATT-3, Attachment 3 Core Spray System Electrical Lineup Checklist, Rev. 20
1-47E814-1, U2 Core Spray System Flow Diagram, Rev. 23
2-OI-75 "Core Spray System"
2-OI-75 Att. 1 "Core Spray System Valve Lineup Checklist"
2-OI-75 Att. 2 "Core Spray System Panel Lineup Checklist"
2-OI-75 Att. 3 "Core Spray System Electrical Lineup Checklist"
2-OI-75 Att. 4 "Core Spray System Instrument Inspection Checklist"
DWG 47E814 "Flow Diagram Core Spray System"
Technical Specifications 3.5.2 "Emergency Core Cooling Systems – Shutdown"
DCN 69842, Replace U3 HPCI Turbine Steam Supply Valve with Equivalent Valve With Different Stem Pitch
Drawing 3-47E812-1, Flow Diagram High Pressure Coolant Injection System, Rev. 61
Drawing 3-47E812-2, Flow Diagram HPCI Oil System, Rev. 6
FSAR Section 6.4.1, High Pressure Coolant Injection System, BFN 23.3
FSAR Section 7.4.3.2, High Pressure Coolant Injection System (HPCI) Control and Instrumentation, BFN 23.3
PER 164399, Response to GL 08-01

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1-47E1847-10, I&C Flow Diagram CA System, Rev. 018
2-45E2690-41, Network Diagram, Rev. 0
2-47E809-2, Flow Diagram Off-Gas System, Rev. 34
3-45E767-5, Wiring Diagram Diesel Generator Schematic Diagram, Rev. 20
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3-47E610-23-1, Mechanical Control Diagram RHR Service water system, Rev. 23
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PER 313398, NRC identified that potential common mode failures not consistently evaluated in design process
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PER 318639, DCN 63290A screening should have denoted a yes answer for Question 2 and potentially requires revision of associated evaluation
PER 319195, DCN 69826A screening did not contain sufficient information as a stand-alone document
PER 319200, DCN 69426A screening did not contain sufficient information as a stand-alone document
PER 319217, Cyber Security Assessment not performed for DCN 69409A per procedure SPP-2.6

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3-SIMI-3B, Reactor Feedwater System Scaling and Setpoint Documents, Rev. 85
SR 311992, PMT Needed to be More Specific
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0-TI-230V, Vibration Program, Rev. 7
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PER 301505
LER 05000296/2010-004-00
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NEDP-20, Conduct of The Engineering Organization, Rev. 15
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ECI-0-000-BKR008, Testing and Troubleshooting of Molded Case Circuit Breakers and Motor Starter Overload Relays, Rev. 92
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EPI-0-000-MOV001, Electrical Preventive Maintenance for Limitorque Motor Operated Valves, Rev. 53
EPI-0-000-MCC001, Maintenance and Inspection of 480VAC and 250VDC Motor Control Centers, Rev. 71
MCI-0-74-VLV002, Residual Heat Removal Motor Operated Valves FCV-74-47, -48, -53, and -67 Disassembly, Inspection, Rework, and Reassembly, Rev.14
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WO 09-727665-000, JOG Replace 74-67 Disc and Relap Valve Seats
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WO 110689220, PM/Testing/Inspection Limitorque MOV 74-67
WO 10676572, Limitorque PM on 74-67
WO 110689184, Breaker PM 480V RMOV BD 2E Compt 2C

Section 1R20: Refueling and Other Outage Activities
3-GOI-100-1A, Unit Startup, Rev. 92
3-GOI-200-2, Primary Containment Initial Entry and Closeout Inspection, Rev. 30
3-OI-47, Turbine-Generator System, Rev. 91
3-SR-3.3.1.1.5, SRM and IRM Overlap Verification, Rev. 5
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2-OI-74, Residual Heat Removal System, Rev. 156
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Operator Logs, dated March 24 – 25, 2011
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List of Closed Secondary Containment Work Orders, dated February 22, 2010
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Procedures, Guidance Documents, and Manuals
NPG-SPP-05.0, Radiological and Chemistry Control, Rev. 0
NPG-SPP-05.1, Radiological Controls, Rev. 2
RCI-1.1, Radiation Operations Program Implementation, Revision 144
RCI-1.2, Radiation, Contamination and Airborne Surveys, Revision 12
RCI-17, Control of High Radiation Areas and Very High Radiation Areas, Revision 68
RCI-9.1, Radiation Work Permits, Revision 67
RCI-26, Radiation Protection Department Standards and Expectations, Revision 18
RCI-34, Remote Monitoring, Revision 10
RCI-40.0, RP Actions for Operation's Unit 0 (Common) Procedural Hold Points, Revision 11
RCI-40.1, RP Actions for Operation's Unit 1 Procedural Hold Points, Revision 16
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RCI-40.3, RP Actions for Operation's Unit 3 Procedural Hold Points, Revision 19
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Browns Ferry Technical Specification 5.7 Administrative Controls-High Radiation Area

Records and Data
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(Annual Inventory Of Non-Fuel SNM and Other Items (Trash) In Unit 1, 2 And 3 Spent Fuel Pools Performed 7/14,-15 and 21/ 2010 Respectively.)
Browns ferry nonexempt byproduct and source material source list
Browns Ferry Nuclear Plant Quality Assurance Fleetwide Radiation Protection Organizational Effectiveness Assessment report - QA-BF–10-017, dated 10/14/2010

Radiation Work Permits
RWP 10170025 Unit 1 RFO 8 Reactor Building RP Support
RWP 10180077 Unit 1 RFO 8 Dry Well Rad Waste Support
RWP 10180186 Unit 1 Primary Dry Well Primary System Pressure Test
RWP 10358888 Unit 3 Dry Well Inspection/Repairs at Approximately 20% Power Level
RWP 10358889 Unit 3 Dry Well Initial Dry Well Entry Approximate 20% Power Level
RWP 10380059 Unit 3 Cycle 14 Dry Well Shielding and Insulation Support

Radiation Surveys
Survey # 122210-10, M0464 ISFSI Pad, 12/22/2010
Survey # 102010-06, M0464 ISFSI Pad, 10/20/2010
Survey # 021010-15, M0464 ISFSI Pad, 2/10/2010
Survey # 012011-15, Unit 1 RXB 565 DW Access, 1/20/2011
Survey #102910-38, Unit 1 RXB 565' Tip Room, 10/29/2010
Survey # 081309-3, Unit 2 RXB 565' Tip Room, 8/13/2009
Survey # 072307-3, Unit 3 RXB 565' Tip Room, 7/23/2007

Corrective Action Program (CAP) Documents
PER 215340 RX1 North East Quad Was Posted As A HRA Due To Elevated Dose Rates On The RWCU Resin Line In The Overhead.
PER 221481 This SR Is To Document a Trend PER For Workers Failing to Recognize and Obey High Radiation Area (HRA) postings.
PER 139212 Nuclear Security Officer Violated a High Radiation Area, Contaminated Area Boundary (Excluded from RCA Required RPM, PM and Site VP Signatures To Get Back In.)
PER 221112 Individual Entered a Posted High Radiation Area on the Wrong RWP
PER 222492 Worker Fails to Follow RP instructions
PER 222888 A Radiation Protection Tech Intercepts Two AUOs Attempting to Enter DW exclusion area.
PER 151501Two AUOs Failed To Log Off Locked High Rad RWP After Exiting Unit 3 DW.
PER 246807 Individual Entered the RCA On The Wrong RWP
PER 330414 Byproduct Material Was Identified At The Bellefonte In-Processing Facility at Hollywood, AL.

Section 2RS2: Occupational ALARA Planning and Controls
Procedures, Guidance Documents and Manuals
Tennessee Valley Authority Browns Ferry Nuclear Power Plant Long-Term Collective Radiation Exposure Reduction Plan 2010-2015
TVA Browns Ferry Nuclear Plant Fiscal Year 2009 Annual ALARA Report
RCI-9.1 BFN Radiation Work Permits, Rev 67
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ALARA Planning Report 11-0046, U2R16 Outage- Snubber Maintenance
ALARA Planning Report 11-0041, U2R16 Outage- Refuel Floor Activities

ALARA Planning Report 11-0042, U2R16 Outage- Drywell Scaffold Support
ALARA Planning Report 11-0049, U2R16 Outage- “A” & “C” RHR Motor Replacements
ALARA Planning Report 11-0048, U2R16 Outage- Remove/Test/Reinstall MSRV’s
ALARA Planning Report 11-0031, U2R16 Outage- Maintenance on the RHR/Core Spray/ RCIC/ HPCI/ MS Valves to meet JOG Requirements
ALARA Planning Report 10-0020, U1R8 Outage- Scaffolding Support
ALARA Planning Report- Post Job Report 10-0020- U1R8 Outage scaffolding support
ALARA Planning Report 10-0032, U1R8 Outage- Refuel floor outage support
ALARA Planning Report- Post Job Report 10-0032, U1R8 Outage-Refuel floor outage support

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11280167 Rev 0, Unit 2 Drywell All Elevations
11280165 Rev 0, Unit 2 Drywell All Elevations
11280166 Rev 0, Unit 2 Drywell All Elevations
11280115 Rev 0, Unit 2 Drywell All Elevations
11280116 Rev 0, Unit 2 Drywell All Elevations
11270215 Rev 0, Unit 2 Reactor Building All Elevations
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11290001 Rev 0, Unit 2 Reactor Building Elevation 664
11280045 Rev 0, Unit 2 Drywell All Elevations
11280046 Rev 0, Unit 2 Drywell All Elevations

Radiological Surveys
030111-17, M0249 Unit 2 RXB 565’ General Area
022811-2, M0184 Unit 2 RXB 551’ Top of Torus
012911-5, M0297 Unit 2 RXB 541’ SW Quad
021611-9, M0082 Unit 2 RXB 519’SW Quad
022311-27, M0082 Unit 2 RXB 519’ SW Quad

CAP Documents
PER 239813- MMG Shop ALARA estimate 248% under dose received WW1027
PER 223264- Bad planning causes the receiving of uncalled for and wasted dose, again.
PER 222829- Bad planning causes the receiving of uncalled for and wasted dose.
PER 320404- While performing 1-SR-3.5.1.1, I&C exceeded dose goal due to procedure non-compliance
PER 244600- High Radiation gates are increasing dose during AUO rounds
PER 221989- An RSC meeting was convened to review the outage dose goal for revision

Section 2RS3: In-Plant Airborne Radioactivity Control and Mitigation
Procedures, Guidance Documents, and Manuals
RCI-3.1, “Respiratory Protection Program Implementation”, Rev. 33
NPG-SPP-05.10, “Radiological Respiratory Protection Program”, Rev. 0
EPIP-12, “Emergency Equipment and Supplies”, Rev. 8
FP-0-049-CMP001, “Operation of Breathing Air Compressors”, Rev. 5
RCI-11.2, “Radiation Protection Airborne Instrument Maintenance”, Rev. 2
RCTP-101, “Operation of the Mask Fit System”, Rev. 1
NPG-SPP-03.1, “Corrective Action Program”, Rev. 1

Records and Data
SCBA Air Compressor Gas Quality Testing Results, 12/9/10
Low Pressure Supplied Air Compressor Gas Quality Testing Results, 10/5/09, 12/16/09, 2/15/10, 2/26/10, 6/4/10, 10/12/10, 12/3/10, 2/7/11
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TEDE-ALARA Evaluation 11-34, Breach Valve 2-FCV-74-53/67/57/61
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MSA Authorized Repair Center Certificates, 9/30/10
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CAP Documents
QA-BF-10-017, RP Organizational Effectiveness Assessment
SR 332590, Some SCBA units have exceeded grace period for annual flow testing
SR 332190, Some respirator filters found to be obsolete per NIOSH
SR 330892, Fewer than required spare air bottles stored at 557’ location in turbine building
PER 318057, Improper storage of respirators

Section 2RS4: Occupational Dose Assessment
Procedures and Guidance Documents
RCI-8.1, “Internal Dosimetry Program Implementation”, Rev. 42
RCI-47, “Diving Operations in the Radiologically Controlled Area”, Rev. 0
RCDP-7, “Bioassay and Internal Dose Program”, Rev. 3
RCDP-10, “Personnel Contamination Reporting”, Rev. 4
NPG-SPP-05.1, “Radiological Controls”, Rev. 2
RCTP-106, “Special Dosimetry Operations”, Rev. 1
NPG-SPP-03.1, “Corrective Action Program”, Rev. 1

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NVLAP Personnel Dosimetry Performance Testing, 4th Quarter 2010, TVA, Chattanooga, TN
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RWP 11270138, U2R16 Outage Torus/CST Underwater Inspections, Rev. 0
Radiological Survey 040205-10, U2 Rx Bldg 541’ Inside Torus
PCE Logs and Selected Investigation Paperwork, 1/1/10 – 12/31/10

CAP Documents
QA-BF-10-017, RP Organizational Effectiveness Assessment
QA-CH-09-004, Assessment of Environmental Radiological Monitoring and Instrumentation Program, Western Area Radiological Laboratory
SR 137994, Internal Contamination/Positive Whole Body Count

SR 331484, RCI-8.1 provides instructions for routine in-vivo bioassay in the appendix that describes in-vitro sampling techniques
PER 219627, Investigative WBC detected intake of Co-60
PER 229667, Discrepancies between TLD and ED readings > 25%
PER 216701, TLD sent through security x-ray equipment
PER 223427, Worker lost TLD

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Procedures
CI-138, Reporting NEI Indicators
NPG-SPP-02.2, Performance Indicator Program, Rev. 0
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1-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Rev. 0
2-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Rev. 5
3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Rev. 4
U1 Reactor Coolant DEI Chemistry Report dated 3/23/11
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1-SR-2, Instrument Checks and Observations, Rev. 23
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PER 338020
1-SR-3.4.4.1, Manual Calculation of Unidentified, Identified, and Total Leakage, Rev. 04
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Section 4OA2: Identification and Resolution of Problems
PER 292396, Evaluate recent HU cross cutting violations relative to PER 228347
PER 248347, Emerging Trend in H.2.c
Root Cause Analysis Report PER 228347
PER 215591, Potential substantive cross-cutting issue in PI&R
PER 302263, Comp measure for each Dept. working with most difficult procedures
NPG-SPP-03.1, Corrective Action Program, Rev. 1
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NPG-SPP-06.2, Preventive Maintenance, Rev. 3
NPG-SPP-09.18, Integrated Equipment Reliability Program, Rev. 1
NPG-SPP-09.18.1, System Vulnerability Review Process, Rev. 1
NPG-SPP-09.18.2, Equipment Reliability Classification, Rev. 1
NPG-SPP-09.18.3, Equipment Reliability Program ER Strategy Development and Implementation Process, Rev. 0
NPG-SPP-09.18.5, Development of Life Cycle Management Plans, Rev. 0

NPG-SPP-09.18 .7, Single Point Vulnerability Review Process, Rev. 0
NEPD-22, Functional Evaluations, Rev. 9
PER 220336, U3 PCIS GE CR120 Relays Scope Deleted From U3R14
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Change Control Board (CCB) Meeting Minutes, dated 2/08/2011
NCO 910037001, Commitment Closure Summary
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PER 305861, Filter Replacements Going Late
Functional Evaluation, Diesel Generator Lube Oil Filter Replacement, Revisions 1, 2 and 3.
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WO 111330842, D/G B
WO 111330860, D/G C
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PER 331104, Questions concerning FE
PER 328682, D/G Mission Time
PER 332999, Interim Measures Not Performed As Required Per FE
BFN-VTD-E147-0020, Electro-Motive 645E4 Turbocharged Engine Manual Rev. 0
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WO# 112075021
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Predictive Maintenance Report, Vibration Analysis Assessment 2A RHR Motor, 3/23/11
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2-SI-4.5.C.1(3-COMP), RHRSW Comprehensive Pump and Header Test, Rev. 18
2-SI-4.5.C.1(2), EECW Pump Operation, Rev. 113
2-SI-4.5.C.1(2-COMP), EECW Comprehensive Pump Test, Rev. 112
PER 146452, Pump Flow Test Instrumentation Meeting ASME Code Requirements
PER 156818, RHRSW Pump B2 Failure
PER 175244, 0-TI-345 Inadequacies
PER 175252, EECW Pump Data Acquisition and Evaluation
PER 175254, River Water Temperature Affects on Pump Performance
PER 175255, Instrumentation Code Relief
PER 213180, EECW Operability With Nonconformances
PER 225844, Untimely Resolution of Non-Conforming ASME Code IST Test Condition
Technical Specifications 5.5.6 Inservice Testing Program, Amendment 239
0-TI-278, Minimizing Primary Coolant Sources Outside Containment
NPG-SPP-06.8, Leak Reduction Program
PER 300003, RHR Leak Outside Containment
PER 317464, TS 5.5.2 Primary Coolant Leaks Outside Containment
EWR 11PROG999053, Current ECCS Leakage Calculation
0-TI-560, Control Room Habitability Program

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Section 4OA3: Event Follow-up
Apparent Cause Evaluation (ACE) for PERs 219150 and 242068, Rev. 1
PER 242068, FE for PER 219150 concluded nonconformance
PER 285603, Operability and functionality determination weaknesses
PER 259309, 1-PIS-64-56B trending close to the max deviation limit
Functional Evaluation (FE) for PER 219150, Rev. 1
1-SR-2, Instrument Checks and Observations, Rev. 23
1-47E610-64-1, Mechanical Control Diagram Primary Containment System, Rev. 64
1-47E600-57A, Mechanical Instruments and Controls, Rev. 01
1-47W600-57 Mechanical Instruments and Controls, Rev. 00
N1E-003, Instrument and Instrument Line Installation and Inspection, Rev. 07
SR 331663, LER 50-259/2010-002-00 contains typographical error
PER 301505, Manual reactor scram on U3
LER 05000296/2010-004-00
NEDP-20, Conduct of The Engineering Organization, Rev. 12
NEDP-20, Conduct of The Engineering Organization, Rev. 15
BFN System Monitoring Plan for the raw cooling water system

Section 4OA5: Other: Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)
Records and Data
Nonexempt Byproduct and Source Material Source List
Email: NSTS Annual Inventory Reconciliation Acknowledgement for License Number DPR-33, 1/24/2011

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