UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 28, 2006

Tennessee Valley Authority
ATTN.: Mr. K. W. Singer
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2006002, 05000260/2006002, AND 05000296/2006002

Dear Mr. Singer:

On March 31, 2006, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your operating Browns Ferry Unit 2 and 3 reactor facilities. The enclosed integrated quarterly inspection report documents the inspection results, which were discussed on April 6, 2006, with Mr. J. DeDomenico and other members of your staff.

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

Additionally, the enclosed report also documents some inspection of Unit 1 that was performed per our letter to you on December 29, 2004, regarding the transition of Unit 1 into the Reactor Oversight Program (ROP). In that letter we indicated that the NRC had determined that the ROP cornerstones of Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection would be incorporated into the routine ROP baseline inspection program effective January 1, 2005. Remaining results from our inspection of your Unit 1 Recovery Project continue to be documented in a separate Unit 1 integrated inspection report.

This report documents two NRC-identified findings and a self-revealing finding of very low safety significance. Two of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because the findings were entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any non-cited violation or finding in the enclosed report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at the Browns Ferry Nuclear Plant.

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

Sincerely,

/RA/

Malcolm T. Widmann, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000259/2006002, 05000260/2006002 and 05000296/2006002
w/Attachment: Supplemental Information

cc w/encl.: (See page 3)

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Report to Karl W. Singer from Malcolm T. Widmann dated April 28, 2006.

SUBJECT: BROWNS FERRY NUCLEAR PLANT - INTEGRATED INSPECTION REPORT 05000259/2006002, 05000260/2006002 and 05000296/2006002

Distribution w/encl.:
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U.S. NUCLEAR REGULATORY COMMISSION
REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report Nos.:	05000259/2006-002, 05000260/2006-002, 05000296/2006-002
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 - March 31, 2006
Inspectors:	T. Ross, Senior Resident Inspector R. Monk, Resident Inspector E. Christnot, Resident Inspector C. Stancil, Resident Inspector R. Taylor, Reactor Inspector (Sections 1R02, 1R17) E. Michel, Reactor Inspector (Sections 1R02, 1R17) K. Harper, Reactor Inspector (Sections 1R02, 1R17) R. Rodriguez, Reactor Inspector (Sections 1R02, 1R17) W. Lewis, Reactor Inspector (Sections 1R02, 1R17) J. Rivera-Ortiz, Reactor Inspector (Section 1R08) F. Ehrhardt, Operator License Examiner (1R11.1) R. Hamilton, Senior Health Physicist (2OS3 and 4OA1) A. Nielsen, Health Physicist (2PS1 and 4OA7) J. Griffis, Health Physicist (2PS3 and 4OA5) W. Loo, Senior Health Physicist (Section 2OS1)
Approved by:	Malcolm T. Widmann, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000259/2006002, 05000260/2006002, 05000296/2006002; 01/01/2006 - 03/31/2006; Browns Ferry Nuclear Plant, Units 1, 2, and 3; Maintenance Risk Assessments, Emergent Work Evaluation, and Event Followup.

The report covered a three-month period of routine inspections by the resident inspectors, 6 reactor inspectors, an operator license examiner, two health physicists and two senior health physicists from Region II. One non-cited violation and one finding, of very low safety significance (Green), and one Severity Level IV non-cited violation were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or 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 3, dated July 2000.

A. <u>NRC-Identified and Self-Revealing Findings</u>

- A Severity Level IV non-cited violation (NCV) of 10 CFR 50.73(a)(2)(v)(D) and (vii)(D) was identified by the inspectors for the licensee’s failure to submit a licensee event report for a safety system functional failure of the Unit 2 residual heat removal pressure suppression chamber containment isolation valves. This issue was documented in the licensee’s corrective action program as Problem Evaluation Report 99193.

  In Section IV of the NRC Enforcement Policy, the significance of violations involving the failure to make required reports is not dispositioned using the Reactor Oversight Program’s Significance Determination Process. The licensee’s failure to provide a written event report does potentially impact the NRC's ability to carry out its regulatory function. However, because this failure to report per 10 CFR 50.73 did not actually impede or influence regulatory action, and the condition that required reporting under 10 CFR 50.73 was previously determined to be of very low safety significance in inspection report 05000260/2005003, the NRC has characterized the significance of this reporting violation as a Severity Level IV in accordance with Section IV.A.3 and Supplement I of the NRC Enforcement Policy. (Section 40A3.2)

Cornerstone: Initiating Events

- A Green self-revealing Finding (FIN) was identified for inadequate work instructions and poor work practices associated with maintenance on the 2C reactor feedwater pump that resulted in a Unit 2 reactor trip. This issue was documented in the licensee’s corrective action program as Problem Evaluation Report 87178.

  This finding is greater than minor because it involved human error and inadequate work instructions that affected the human performance and procedure quality attributes of the Initiating Event Cornerstone to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was determined to be of very low safety significance because all safety-related mitigating systems operated as designed during and following the scram. (Section 4OA3.3)

Cornerstone: Mitigating Systems

- A Green non-cited violation (NCV) of 10 CFR 50.65(a)(4) was identified by the inspectors for the licensee's failure to conduct an adequate risk assessment of the Unit 2 systems, and Unit 3 systems affecting Unit 2, that were taken out of service for scheduled maintenance from March 1 through 3, 2006. This resulted in an unrecognized increase in the level of risk as determined by a probabilistic safety analysis (PSA) evaluation by the licensee. This issue was documented in the licensee's corrective action program as Problem Evaluation Report 98414.

  This finding is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the cornerstone objective in that the licensee failed to perform an adequate risk assessment prior to conducting online maintenance. The licensee's risk assessment did not consider all the risk significant systems and support systems that were out of service which, when properly evaluated, did result in an increased level of risk from a PSA perspective. However, the finding was of very low safety significance because the risk deficit for Incremental Core Damage Probability was less than 5E-6 and for Incremental Large Early Release Probability was less than 5E-7, and at least two risk management actions were in place. This finding involved the cross cutting aspect of Human Performance for failure to recognize and follow established procedures for adequately assessing the risk associated with online maintenance. (Section 1R13)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 was defueled and in a recovery status for the entire report period.

Unit 2 operated at essentially full power for the entire report period, except for several downpowers. On January 28, 2006, unit power was reduced to 70% to repair a steam leak on the extraction steam supply to a main feedwater (MFW) heater; unit power returned to 100% on January 30. On February 1, unit power was reduced to 79% due to unexpected isolation of the main steam (MS) extraction supply lines to the 2C1 and 2C2 MFW heaters; unit power returned to 100% on February 2. On February 16, unit power was reduced to 57% to remove the 2B Variable Frequency Drive (VFD) to repair a coolant leak; unit was returned to 100% on February 17.

Unit 3 began the inspection period operating at essentially full power, but shutdown January 15, 2006 due to failure of the second seal on the 3B Recirculation pump. Both seals on both recirculation pumps were replaced. The unit was restarted on January 20 and achieved full power on January 25. Unit 3 continued to operate at 100% power until it was shutdown on February 28 for its twelfth refueling outage (U3C12). After the outage was complete, the unit was restarted and placed online on March 22. Low power testing and power ascension to 100% were completed on March 26. On March 27, power was reduced to 90% due to unexpected loss of the 3C2 MFW heater. The unit was returned to full power the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility, Updated Final Safety Analysis Report (UFSAR), or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for **eight** changes and additional information, such as calculations, supporting analyses, the UFSAR, and drawings, to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The **eight** evaluations reviewed are listed in the report Attachment.

The inspectors also 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 10 CFR 50.59. The **15** "screened out" changes reviewed are listed in the report Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

Partial System Walkdown. The inspectors performed partial walkdowns of the three safety systems listed below to verify train operability, as required by the plant Technical Specifications (TS), while the other redundant trains were out of service or after the specific safety system was returned to service following maintenance. These inspections included reviews of applicable TS, applicable operating instructions (OI), and/or piping and instrumentation drawings (P&IDs), which were compared with observed equipment configurations to identify any discrepancies that could affect operability of the redundant train or backup system. The systems selected for walkdown were also chosen due to their relative risk significance from a Probabilistic Safety Assessment (PSA) perspective for the existing plant equipment configuration. The inspectors verified that selected breaker, valve position, and support equipment were in the correct position for system operation.

- Unit 3 Residual Heat Removal (RHR) System (Division II) per PI&D flow diagram 3-47E811
- Units 1 and 2 B Emergency Diesel Generator (EDG) per PI&D flow and control diagrams 0-47E861-1 and 0-47E861-6
- Unit 2 RHR System (Division II) per PI&D flow diagram 2-47E811

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 2 Standby Liquid Control (SLC) System, using the applicable P&ID flow diagram 2-47E854-1 and 2-OI-63, to walkdown and verify equipment alignment 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 control room operator logs, the applicable System Health Report, and any Problem Evaluation Reports (PERs) that could affect system alignment and operability for the past year.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Walkdowns. The inspectors reviewed licensee procedures, Standard Program and Process (SPP)-10.10, Control of Transient Combustibles, and SPP-10.9, Control of Fire Protection Impairments, and conducted a walkdown of the ten fire areas (FA) and fire zones (FZ) listed below. Selected fire areas/zones 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 equipment or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure SPP-10.9. 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 fire fighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, were in place.

- Unit 3 Battery and Battery Board (FA-19)
- Intake Structure (FA-25)
- Unit 2 Reactor Building - West side (FZ 2-1)
- 3A 480V Shutdown Board Room (FA-14)
- 3B 480V Shutdown Board Room (FA-15)
- Unit 3 “A” Electric Board Room (FA-13)
- Unit 2 Reactor Building - East side (FZ 2-2)
- Unit 3 4KV Shutdown Board (Division 1) Room (FA-22)
- Unit 3 4KV Shutdown Board (Division II) Room (FA-23)
- Unit 3 4KV Shutdown Board Inter-tie Board Room (FA-24)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed a review of the Unit 2 RHR and Core Spray (CS) pump rooms and Under-Torus area for internal flood protection measures. 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 following documents: UFSAR; Design Criteria Browns Ferry Nuclear (BFN)-50-C-7105, Internal Flooding Design Basis; Emergency Operating Instruction 3, Secondary Containment Control; and, Browns Ferry Unit 2 Individual Plant Examination, Browns Ferry Internal Floods

Analysis. Furthermore, the inspectors reviewed the Browns Ferry Nuclear Plant Probabilistic Safety Assessment Initiating Event Notebook, Initiating Event Frequencies, for licensee commitments.

The inspectors performed walkdowns of risk-significant areas, susceptible systems and equipment, including the Unit 3 RHR, CS pump rooms, High Pressure Coolant Injection (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. Plant procedures for mitigating flooding events were also reviewed to verify that licensee actions were consistent with the plant's design basis assumptions.

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 completed preventive maintenance procedures, work orders, and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems Inservice Inspection

a. Inspection Scope

On March 6-10, 2006, the inspectors reviewed the implementation of the licensee's Inservice Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries. The inspectors selected a sample of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI required examinations and a sample of Risk Informed ISI Program examinations.

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate compliance with TS, ASME Section XI, and ASME Section V requirements (1995 Edition/1996 Addenda for examinations credited to the third period of the second 10-year ISI interval, and 2001 Edition/2003 Addenda for examinations credited to the first period of the third 10-year ISI interval), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI, IWB-3000 or IWC-3000 acceptance standards.

Specifically, the inspectors reviewed NDE reports of Visual (VT) examinations, Magnetic Particle (MT) examinations, and Radiographic (RT) examinations as described below:

VT Report	Component
R-017 (VT-3)	3-47B465-503, Snubber in Reactor Water Recirculation (RWR) System (Category F-A)
R-018 (VT-3)	3-47B400-82, Rigid Hanger on Main Steam (MS) System (Category F-A)
R-013 (VT-3)	3-47B465-513, Piping Support in RWR System (Category F-A)
R-019 (VT-3)	3-47B465-501, Piping Support in RWR System (Category F-A)

Recorded VT Video	Component
EVT-1 on Nozzle N5-A	Inner Radius of Reactor Vessel to Core Spray Nozzle N5A Weld (ASME Class 1)

MT Report	Component
R-025	3-47B400-115-1A, Welded Attachment to Pipe in MS System (Category B-K)
R-026	3-47B400-116-1A, Welded Attachment to Pipe in MS System (Category B-K)
R-044	3-47B400-114-1A, Welded Attachment to Pipe in MS System (Category B-K)

RT Report and Films	Component
WO-04-720059-000	Weld Number (No.): MS-3-002-022, Weld in a MS System Valve (ASME Class 1)

In addition, the inspectors reviewed the NDE report and directly observed the calibration and execution of the manual ultrasonic (UT) examination of the following welds:

- Weld No.: N4D-NV, Reactor Vessel to Feedwater Nozzle Weld, ASME Class 1 (Associated with Problem Evaluation Report 96089)

- Weld No.: DSAS-3-03, 6-inch Line in MS System, ASME Class 2

The inspectors reviewed UT report R-031, which describes a successive examination for a previous recordable indication in weld GR-3-63 (valve to pipe weld in RWR system, ASME Class 1), to determine if the evaluation and disposition of the indication was in accordance with the applicable version of ASME Section XI, IWB-3000.

Qualification and certification records for examiners, inspection equipment, and consumables, along with the applicable NDE procedures for the above ISI examination activities, were reviewed and compared to the requirements stated in ASME Section V and Section XI.

The inspectors also reviewed a sample of welding activities performed since the beginning of the last refueling outage for ASME Class 1 and 2 piping. The inspectors reviewed welding procedures, procedure qualification records, welder qualification records, and NDE reports for the following welds:

- Weld No.: RWR-3-007, 009, 010, and 011, 2-inch Drain Line Valve in RWR System, ASME Class 1

- Weld No.: HPCI-3-020-001, 002, Replacement of 2-inch Check Valve 3-CKV-073-0629 in HPCI System, ASME Class 2

- Weld No.: MS-3-002-021, Installation of 3-inch valve FCV-1-056 in MS System, ASME Class 1 (Note: NDE report not available for review at the time of the inspection.)

The inspectors conducted a containment walk-down of multiple drywell elevations to assess, in general, the material condition of structures, systems, and components.

Furthermore, the inspectors reviewed the licensee's operating experience assessment for issues associated with NRC Information Notice 2006-01, "Torus Cracking in a BWR Mark I Containment."

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the corrective action program as PERs. The inspectors reviewed the PERs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. In addition, as indicated above, the inspectors independently verified the implementation of corrective actions for an ISI issue documented in PER 96089. The inspectors performed this review 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 report Attachment.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Internal Inspections

a. Inspection Scope

The inspectors reviewed NDE activities associated with the inspection of Reactor Vessel internal components (Boiling Water Reactors Vessel Internals Project). The inspectors verified that inspection activities were conducted in accordance with established procedures and that indications, if present, were identified and properly documented in the examination reports. Specifically, the inspectors reviewed portions of electronic data and recorded videos for the following activities:

- VT-1 of Adjusting Screws in Jet Pumps 9, 10, and 14
- VT-1 of Wedge WD-1 in Jet Pump 4
- VT-1 of Core Spray T-box (Nozzle N5A, 120° azimuth)
- VT-1 of Steam Dryer Bank 4 Vertical Welds
- VT-1 of Steam Dryer Tie Bar (TB)-1/2-02
- Automated UT of Core Shroud Welds H1, H2, H4, and H6

b. Findings

No findings of significance were identified.

.3 Flow Assisted Corrosion Program

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's Flow Assisted Corrosion (FAC) program to verify that systems subject to FAC were included within the scope of the program. For two FAC reports, the inspectors reviewed the UT data, data analysis using "FAC Manager," sample expansion, and program recommendations for unsatisfactory results. The reports reviewed by the inspectors are listed in the report Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Annual Operating Test Results

a. Inspection Scope

Annual review of Licensee Requalification Examination Results. On January 19, 2006, the licensee completed the comprehensive requalification biennial written examinations and annual operating tests required to be given to all licensed operators by 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the written examinations, individual operating tests, and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

.2 Requalification Activities Review

a. Inspection Scope

On February 6, 2006, the inspectors observed two operating crews during a simulator as-found evaluation per Simulator Evaluation Guides OPL 177.083 and 178.083 to verify that crew performance was in accordance with licensee procedures and regulatory requirements. Although the scenarios were identical, one evaluation was conducted on the Unit 2 simulator and the other was conducted on the Unit 3 simulator.

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

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of Abnormal Operating Instructions (AOI), Emergency Operating Instructions (EOI) and Operational Contingencies
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedures
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the Unit Supervisor and Shift Manager

The inspector also attended the post-exam critique to assess the effectiveness of the licensee evaluators, and to verify that licensee-identified issues were comparable to issues identified by the inspector.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

.1 Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two systems listed below, which exceeded their performance criteria, with regard to some or all of the following attributes: (1) work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2), (8) system classification in accordance with 10 CFR 50.65(a)(1); and (9) appropriateness and adequacy of (a)(1) goals and corrective actions (i.e., Ten Point Plan). The inspectors also compared the licensee's performance against site procedure SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; and SPP 3.1, Corrective Action Program. The inspectors also reviewed applicable work orders, 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.

- Unit 2 and 3 Drywell Head repetitive Local Leak Rate Test failures
- 1B Control Rod Drive (CRD) pump excessive unavailability

b. Findings

No findings of significance were identified

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

u. Inspection Scope

For planned online work and/or emergent work that affected the seven risk significant systems listed below, the inspectors reviewed licensee maintenance risk assessments and actions taken to plan and control work activities to effectively manage and minimize risk. The inspectors verified that risk assessments and risk management actions (RMA) were being conducted as required by 10 CFR 50.65(a)(4) and applicable procedures such as SPP-6.1, Work Order Process Initiation, SPP-7.1, Work Control Process and 0-TI-367, BFN Dual Unit Maintenance Matrix. The inspectors also evaluated the adequacy of the licensee's risk assessments and the implementation of RMAs.

- 3C EDG, 3D RHR Pump, D2 Residual Heat Removal Service Water (RHRSW) Pump, and Maury 500 KV Line out-of-service (OOS) (1/5)
- Work activities during Unit 3 forced outage that affected Unit 2 (1/17)
- 3D 480V Reactor Motor-Operated Valve (RMOV) Board and 3EB Shutdown Board Battery OOS (1/24)
- 1B CRD Pump, Unit 2 Reactor Core Isolation Cooling pump, 2A RHR Pump/Hx, and D2 RHRSW Pump OOS (1/26)
- 3A RHR Pump, 3D RHR heat exchanger and 3B Control Bay Chiller OOS (2/9)
- 3B and 3D RHR Pumps, 3B Reactor Feedwater Pump, 3B Condensate Booster Pump and 3B Condensate pump OOS (2/22)
- Units 1 and 2 "B" EDG OOS, during U3C12 Division I equipment outage which included 3A and 3B EDGs, and associated 3EA and 3EC 4KV Shutdown Boards (3/1 thru 3/3)

b. Findings

Introduction: A Green non-cited violation (NCV) of 10 CFR 50.65(a)(4) was identified by the inspectors for the licensee's failure to conduct an adequate risk assessment of the Unit 2 systems, and Unit 3 systems affecting Unit 2, that were taken OOS from March 1 through 3, 2006. This resulted in an unrecognized increase in the level of risk as determined by a probabilistic safety analysis (PSA) evaluation by the licensee.

Description: On March 2, 2006, the inspectors examined risk significant work activities conducted during the U3C12 Division I equipment outage that affected Unit 2, particularly while the B EDG was OOS for scheduled maintenance from March 1 to March 3. The licensee had concluded that the risk associated with this maintenance was acceptable per their risk assessment tools "BFN Dual Unit Maintenance Matrix" and Sentinel. However, the inspectors subsequently determined that the licensee had not adequately assessed the level of risk with regard to the operating unit (Unit 2) and had not properly aligned the RHRSW pumps to minimize risk. While the A3 and C3 Emergency Equipment Cooling Water (EECW) pumps were OOS, the swing C1 RHRSW was aligned to replace the C3 EECW pump. But since the B EDG supports the C1 RHRSW pump, the C1 pump was also OOS and the licensee should have aligned the swing A1 RHRSW pump to replace the A3 EECW pump per 0-TI-367.

The scope of equipment taken OOS (see last bullet above) exceeded that which was allowed by the licensee's BFN Dual Unit Maintenance Matrix required by SPP-7.1 and TI-367 for screening risk significant work activities. Specifically, the Matrix could not be used in situations where more than two systems were OOS. Per these procedures, this condition would have required the licensee to conduct a detailed PSA evaluation. The adverse affect on risk in removing Unit 3 EDGs from service, with respect to the dual unit matrix for Unit 2, was not recognized by responsible Work Week Management or Operations personnel. Furthermore, the licensee's Sentinel program was not properly modeled for considering the impact of Unit 3 EDGs on Unit 2. Sentinel indicated that the risk Matrix was satisfied (i.e. Green), when it should have indicated the risk Matrix was not satisfied (i.e. Orange). This condition went unrecognized until identified by the inspectors.

Once this condition was brought to the licensee's attention, they promptly reconfigured the A1 RHRSW pump to replace the A3 EECW pump. The licensee also conducted a detailed PSA evaluation that initially concluded the level of increased risk was acceptable. After reviewing the initial PSA evaluation, the inspectors determined that the licensee failed to consider all of the risk significant equipment that was OOS. Consequently, the licensee re-performed their PSA evaluation. On March 16, the results of this evaluation indicated that the Incremental Core Damage Probability (ICDP) had increased to 4.5E-06 and the Incremental Large Early Release Probability (ILERP) had increased to 1.47E-07 for the scheduled three days of maintenance. Both of these values exceeded the Yellow risk threshold specified in SPP-7.1 and required the implementation of risk management actions (RMA).

Prior to the U3C12 refueling outage, the licensee had reviewed the risk impact on Unit 3 of removing the 3A and 3B EDGs from service from an outage risk perspective using their Outage Risk Assessment and Management (ORAM) tool. During the period of time from March 1 to March 3, the level of risk for "onsite power" due to the unavailable 3A and 3B EDGs was only considered to be Yellow by ORAM. However, the overall level of outage risk for Unit 3 was considered to be Orange according to ORAM due to "offsite power" unavailability during the same time. Consequently, the overall plant risk per the licensee's new procedure BP-336, Risk Determination and Risk Management, was considered to be Orange due to ORAM being Orange even though Unit 2 Sentinel was Green. Because the overall plant risk was considered to be Orange, the licensee did put in place certain RMAs, such as increased awareness, and measures to ensure redundant trains (i.e., 3C, 3D, A, C, and D EDGs) were protected.

Analysis: The inspectors determined that the licensee's failure to perform an adequate risk assessment was more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the cornerstone objective. The licensee's risk assessment did not consider all the risk significant systems and support systems that were OOS which, when properly evaluated, resulted in an increased level of risk for Unit 2 (i.e., Green to Yellow) from a PSA perspective. The inspectors assessed this finding using the Inspection Manual Chapter 0609, Appendix K, Maintenance Risk Assessment and Risk Management Significance Determination Process, and determined the finding to be of very low safety significance (i.e., Green) per Flowchart 1, Risk Assessment Details. More specifically, the finding was considered Green because the risk deficit for ICDP was less than 5E-6 and for LERP less than 5E-7 LERP, and at least two RMAs were in place. This finding involved the cross-cutting area of Human Performance for failure to appropriately follow established procedures, specifically SPP-7.1 and TI-367, as evidenced by the licensee's failure to recognize that with more than two systems out-of-service the BFN Dual Unit Maintenance Matrix could not be used and thereby required a detailed PSA evaluation.

Enforcement: The regulatory requirement for "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," 10 CFR 50.65(a)(4), states, in part, that the licensee shall assess and manage the increase in risk that may result from proposed maintenance activities. Procedure SPP-7.1 and associated TI-367 implemented the requirements of 10 CFR 50.65 (a)(4) by requiring a risk assessment be performed prior to online maintenance activities. Contrary to the above, on March 1, the licensee failed to recognize that scheduled maintenance on the 3A, 3B, and B EDGs exceeded the capability of the BFN Matrix and according to TI-367 required a detailed PSA evaluation. This evaluation was not performed accurately until March 16, well after the EDGs were returned to service. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as PER 98414, this violation is being treated as an NCV in accordance with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000260/2006002-01, Failure to Perform An Adequate Risk Assessment.

1R14 Operator Performance During Non-Routine Evolutions and Events

a. Inspection Scope

For the non-routine evolutions described below, the inspectors evaluated operator performance through interviews, observations, examining available information (e.g. operator logs, plant computer data, and trend recorders), and reviewing applicable PERs to determine what occurred, how the operators responded, and to verify that the response was in accordance with plant procedures (e.g., normal operating instructions, annunciator response procedures, AOIs, etc.).

- On January 15, 2006, inspectors were notified by plant management that the #2 seal on the 3B RWR pump had apparently failed. Previously, the #1 seal for this pump had failed and control room operators had been closely monitoring seal leakage. With the failure of the second seal, leakage increased by about 1.6 gpm. Plant management had previously prepared a plan, should both seals fail, to promptly reduce reactor power, remove the 3B RWR pump from service, isolate the 3B pump, and then shutdown the unit. Inspectors reported to the control room to observe shutdown activities. The inspectors observed power reduction in accordance with 3-GOI-1-12A and the removal of the 3B RWR pump in accordance with 3-OI-68, Reactor Recirculation System. Operators were unsuccessful in isolating the pump due to the suction valve 3-FCV-68-77 failing to close. The operators then manually scrammed the reactor per 3-AOI-100-1, Reactor Scram. All systems responded as expected with the exception of PCB 234, main generator output breaker, which failed to open on turbine trip.

- On February 1, inspectors witnessed operator response to the unexpected isolation of the main steam extraction supply lines to the 2C1 and 2C2 High Pressure MFW Heaters. The inspector observed operators reduce power to about 79% in accordance with 2-AOI-6-1-A, High Pressure Feedwater Heater String/Extraction Steam Isolation and 2-GOI-100-12, Power Maneuvering. The inspector also attended the pre-job brief and observed the return to full power the next day in accordance with 2-GOI-100-12, the applicable Reactivity Control Plan (RCP), and associated OIs. The inspector discussed the RCP with both the Unit Supervisor and responsible reactor engineers (RE).

- On February 16, the inspectors attended the pre-job brief for reducing Unit 2 reactor power to approximately 50% and removing the 2B VFD from service due to excessive coolant leaks. The inspectors witnessed operators conduct an orderly down power using RWR pump speed and control rods in accordance with 2-GOI-100-12, 2-OI-68, and the applicable RCP. Per OI-68, Section 7.2, the inspector witnessed operators shutdown the 2B RWR pump and establish single loop operation (SLO). Furthermore, the inspector witnessed RE personnel reset average power range monitor channels for SLO in accordance with 2-SR-3.4.1 (SL), Reactor Recirculation Single Loop Operation. The inspector reviewed and verified licensee compliance with TS, and OI-68 precautions and limitations.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

Routine Baseline Review

a. Inspection Scope

The inspectors reviewed the six operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors reviewed appropriate sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where applicable, the inspectors reviewed licensee procedure SPP-3.1, Corrective Action Program, Appendix D, Guidelines for Degraded/Non-conforming Condition Evaluation and Resolution of Degraded/Non-conforming Conditions, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors reviewed implemented compensatory measures to verify that they worked as stated and that the measures were adequately controlled. The inspectors also reviewed PERs daily to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- 2-PS-67-51, EECW Pressure Switch Setpoint Too Low (PER 95533)
- 3A RHR Room Cooler Motor Drawing Excessive Amps (PER 93311)
- D EDG Potentially Clogged Lube Oil System (PER 93849)
- Unit 3 RHR Loop II, and 3A RHR Pump, Auto-start Circuit Miswired (PER 96868)
- 3-FCV-74-71, RHR Test Line Isolation Valve Failed To Close (PER 97555)
- Unit 2 Drywell Control Air System Piping Undersized (PER 98447)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated engineering change packages for eight modifications, in the Mitigating Systems and Barrier Integrity cornerstone areas, to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. The eight modifications and the associated attributes reviewed are as follows:

EDC 51557, Allow Alternate Disc To Be Used In Residual Heat Removal Service Water System Flow Control Valve (Mitigating Systems)
- Plant Document Updating
- Installation Records

- System Flow Requirements
- Materials/Replacement Components

DCN 60536, Modify The Normal Alignment of the U3 RHR Cross-Tie Loop By Closing BFN-3-SHV-74-150 And Opening BFN-3-FCV-74-46, Rev. A (Mitigating Systems)
- Operations
- Flowpaths
- Process Medium (fluid pressures)
- Licensing Basis

DCN 51643, Modify Valve Internals To Allow For Improved Flow Characteristics, Rev. A (Mitigating Systems)
- Timing (response time)
- Pressure Boundary
- Operations
- Licensing Basis

DCN 62271, Provide Replacement For The 3C DG Air Compressor (Right Bank), Rev. A (Mitigating Systems)
- Energy Needs
- Materials/Replacement Components
- Control Signals
- Flowpaths
- Pressure Boundary

DCN T40676A, Replace GE Magne Blast Breakers With Breakers Which Use Vacuum Bottle Technology, Rev. A (Including PICs 50001A, 50031A, 50059A, 50121A, 50313A, 50427A, 50814A, 50994A, 63176A and FDCNs F40922A, F40949A, F41010, F41218A, F41422A) (Mitigating Systems)
- Energy Needs
- Materials/Replacement Components
- Timing
- Control Signals
- Operations

DCN 65424, Provide Materials Equivalency Replacement For Hancock 5500 Valves Used As Drain Or Root Valves On Units 2, 3 Feedwater Systems, Rev. A (Mitigating Systems)
- Materials/Replacement Components
- Pressure Boundary
- Structural

DCN 63466, Provide Separate Power Supplies To Unit 2 Reactor Feedwater Pump Minimum Flow Bypass Valves Control Circuits (Mitigating Systems)
- Energy Needs
- Materials/Replacement Components
- Process Medium

DCN 60561, Add Shutoff Valve In 3" Portion Of Reactor Head Vent Line (Barrier Integrity)

- Materials/Replacement Components
- Pressure Boundary
- Flowpaths
- Plant Document Updating

For selected modification packages, the inspectors observed the as-built configuration. 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 also reviewed selected PERs and an audit associated with modifications to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the six post-maintenance tests (PMT) listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee’s test procedure to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure 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 or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors also verified that PMT activities were conducted in accordance with applicable procedural requirements, including SPP-6.3, Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors reviewed problems associated with PMTs that were identified and entered into the corrective action program.

- Work Order (WO) 06-710906-000 and 06-710906-000, Inspect and Repair Normal and Alternate Feeder Breakers for 3D RMOV Board
- 2-SI-4.5.C.1(3-COMP), RHRSW Comprehensive Pump and Header Test, Impeller Adjustment of D2 RHRSW Pump
- WO 06-711660-000, 3A RHR Pump Auto-start Circuitry Wiring Discrepancy

- WO 05-717018-001 and 3-SR-3.5.1.8, High Pressure Coolant Injection Main And Booster Pump Set Developed Head and Flow-rate Test, Replacement of HPCI Turbine Remote Trip Solenoid
- WO 03-015955-000 and 03-015955-002, and 3-SR-3.5.3.4, Reactor Core Isolation Cooling (RCIC) System Rated Flow at Low Pressure, Repairs to RCIC Steam Admission Valve
- 2-SR-3.5.1.6(CSII), Core Spray Flow Rate Loop II, Routine Preventative Maintenance of 2B and 2B CS Pumps.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Unit 3 Short Notice Outage

a. Inspection Scope

On January 15 - 20, 2006, the inspectors examined critical activities associated with the Unit 3 RWR seal replacement outage to verify that they were conducted in accordance with TS, applicable procedures, and the licensee's outage risk assessment and management plans. Some of the more significant outage activities observed and/or reviewed by the inspectors were as follows:

- Removal of 3B recirc pump from service
- Reactor cooldown
- Outage risk assessment
- Restart Plant Oversight Review Committee
- Reactor power ascension

The inspectors also verified that selected TS, license conditions, license commitments, and administrative prerequisites were being met prior to Unit 3 mode changes. Furthermore, the inspectors examined RCS identified and unidentified leakage tests.

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required. Resolution and implementation of corrective actions of several PERs were also reviewed for completeness.

b. Findings

No findings of significance were identified.

.2 Unit 3 Scheduled Outage (U3C12)

a. Inspection Scope

During February 28 - March 20, 2006, the inspectors examined critical outage activities to verify that they were conducted in accordance with TS, applicable 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:

Outage Risk Assessment

Prior to the Unit 3 scheduled 20-day refueling outage that began on Feb 28, the inspectors reviewed the Outage Risk Assessment Report and attended outage risk overview meetings 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 defense-in-depth of safety attributes was maintained. The inspectors also reviewed U3C12 Outage Schedule Overview. These reviews were compared to the requirements in licensee procedure SPP-7.2, Outage Management, and TS. These reviews were also done to verify that for identified high risk significant conditions, due to equipment availability and/or system configurations, contingency measures were identified and incorporated into the overall outage and response plan. Furthermore, the inspectors routinely examined the daily U3C12 Refueling Outage Reports and ORAM Safety Function Status Reports. The inspectors frequently discussed posted risk conditions with Operations and outage personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process

On February 28, the inspectors witnessed the shutdown and cooldown of Unit 3 in accordance with licensee procedures SPP-12.1, Conduct of Operations, 3-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations, and 3-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 3-OI-74, Residual Heat Removal System (RHR); 3-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Auxiliary Decay Heat Removal System Failures; and conducted a main control room panel and in-plant walkdowns of system and components to verify correct system alignment. During planned evolutions that resulted in an increased outage risk condition of "Orange" for shutdown cooling, inspectors verified that the plant conditions and systems identified in the risk mitigation strategy were available. In addition, the inspectors reviewed controls implemented to ensure that outage work was not impacting the ability of operators to operate spent fuel pool cooling, RHR shutdown cooling, and/or Auxiliary Decay Heat Removal (ADHR) systems. Furthermore, the inspectors conducted several walkdowns of the ADHR system during operation with the fuel pool gates removed.

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 orders (i.e., TO-2006-001, section 3-085-0001, for CRD hydraulic Unit 3; and TO-2006-001, section 3-074-0013, for RHR Drywell Containment Spray MOV 74-75)
- Verified RCS inventory controls
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) during various modes of operation
- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations and overall integrity
- Examined foreign material exclusion controls particularly in proximity to and around the reactor cavity, equipment pit, and spent fuel pool

Refueling Activities and Containment Closeout

The inspectors witnessed selected fuel handling operations during the Unit 3 reactor core fuel shuffles performed on the refuel floor according to TS and applicable operating procedures, such as 3-GOI-100-3B, Refueling Operations. The inspectors also witnessed and examined the video verification of the final reactor core. Furthermore, the inspectors performed a detailed closeout inspection of the Unit 3 drywell and suppression chamber prior to plant startup and reviewed licensee implementation of 3-GOI-200-2, Drywell Closeout.

Heatup, Mode Transition, Reactor Startup, and Power Ascension Activities

During the week of March 20, the inspectors witnessed portions of the reactor startup, heatup, and power ascension in accordance with 3-GOI-100-1A, Unit Startup and Power Operation, and 3-SR-3.4.9.1(1). The inspectors also verified selected TS, license conditions, license commitments, and administrative prerequisites were being met prior to Unit 3 mode changes. The inspectors also reviewed measured RCS identified and unidentified leakage tests, and verified that containment integrity was properly established. The results of low power physics testing were discussed with RE and Operations personnel to ensure that the core operating limit parameters were consistent with the design.

Corrective Action Program

The inspectors reviewed PERs generated during U3C12 to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required. Resolution and implementation of corrective actions of several PERs were also reviewed for completeness.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions and/or reviewed completed test data of the following eight surveillance tests for the risk-significant and/or safety-related systems listed below to verify that the tests met TS surveillance requirements, UFSAR commitments, and in-service testing (IST) and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the systems, structures, and components were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement.

- 2-SI-4.5.C.1(3-COMP), RHRSW Comprehensive Pump and Header Test
- 2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test [B1 and B2 RHRSW Pumps] *
- 3-SI-4.4.A.1, Standby Liquid Control Pump Functional Test
- 3-SR-3.3.5.1.6 (B II), Functional Testing of RHR Loop II Pump and Minimum Flow Valve Logic
- 3-SR-3.4.5.3, Drywell Floor Drain Sump Flow Integrator Calibration [Unit 3] **
- 3-SR-3.8.1.9(3B OL), Diesel Generator 3B Emergency Load Acceptance Test With Unit 3 Operating
- 2-SR-3.4.5.3, Drywell Floor Drain Sump Flow Integrator Calibration [Unit 2] **
- 3-SR-3.6.1.3.10(B-OUTBD), Primary Containment Local Leak Rate Test Main Steam Line B Outboard: Penetration X-7B ***

* Quarterly IST.
** RCS Leak Detection Test
*** Containment Isolation Valve Leak Rate Test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed licensee procedures 0-TI-405, Plant Modifications and Design Change Control; 0-TI-410, Design Change Control; SPP-9.5, Temporary Alterations; and the temporary modification listed below to ensure that procedure and regulatory requirements were met. The inspectors reviewed the associated 10 CFR 50.59 screening and evaluation and applicable system design bases documentation. The inspectors reviewed selected completed work activities and walked down portions of the systems to verify that installation was consistent with the modification documents.

- Engineering Work Request EWR05CEB001101 for extension of secondary containment beyond the MSIV's to temporary steamline plugs.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Licensee activities for monitoring workers and controlling access to radiologically significant areas were inspected. The inspectors evaluated procedural guidance and directly observed implementation of administrative and physical controls; appraised radiation worker and technician knowledge of, and proficiency in implementing, Radiation Protection (RP) program activities; and assessed worker exposures to radiation and radioactive material.

The inspection focused on the ongoing refueling activities for Unit 3. Radiological postings and material labeling were directly observed during tours of the Unit 1, Unit 2, and Unit 3 turbine and reactor buildings and radwaste processing areas. The inspectors conducted independent surveys in these areas to verify posted radiation levels and to compare with current licensee survey records. During plant tours, control of Locked High Radiation Area (LHRA) keys and the physical status of LHRA doors were examined. In addition, the inspectors observed radiological controls for non-fuel items stored in the spent fuel pool. The inspectors also reviewed selected RP procedures and radiation work permits (RWPs), and discussed current access control program implementation with RP supervisors.

During the inspection, radiological controls for work activities in High Radiation Areas were observed and discussed. The inspectors reviewed the radiological controls associated with the Reactor Building Reactor Water Cleanup Heat Exchanger weld repairs. The inspectors attended pre-job briefings for selected Unit 3 work activities conducted at the Unit 3 drywell control point as workers were briefed of the radiological conditions and RWP requirements by Health Physics Technicians (HPTs). The inspectors observed workers' adherence to RWP guidance and HPTs' proficiency in providing job coverage for various activities in the Unit 3 drywell. Controls for limiting exposure to airborne radioactive material were reviewed and operation of ventilation units and positioning of air samplers were also observed. The inspectors evaluated electronic dosimeter alarm setpoints for consistency with radiological conditions in and around Unit 1 and Unit 3 drywells. In addition, the inspectors interviewed workers in the Unit 1, Unit 2, and Unit 3 reactor buildings to assess knowledge of RWP requirements.

The inspectors evaluated worker exposures through review of data associated with discrete radioactive particle and dispersed skin contamination events. Controls used for monitoring extremity dose and the placement of dosimetry when work involved significant dose gradients were reviewed.

RP program activities were evaluated against 10 CFR Part 20; TS 5.4, Procedures, and TS 5.7, High Radiation Areas; Regulatory Guide (RG) 8.38, Control of Access to High and Very High Radiation Areas in Nuclear Power Plants; and approved licensee procedures. Licensee guidance documents, records, and data reviewed are listed in Section 2OS1 of the report Attachment.

Problem Identification and Resolution. Problem Evaluation Reports associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with RP supervisors. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1, Corrective Action Program, Revision 7. Specific documents reviewed are listed in Section 2OS1 of the report Attachment.

The inspectors completed 21 of the required 21 samples for Inspection Procedure (IP) 71121.01.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope

Radiation Monitors and Protective Equipment. The inspectors reviewed the operability and maintenance of selected radiation detection and respiratory protective equipment. The inspection consisted of document review, discussions with plant personnel, and observation of routine testing for the following items: Area Radiation Monitors (ARMs), Continuous Air Monitors (CAMs), personnel monitors, portable detection instruments, and Self-Contained Breathing Apparatus (SCBA).

The inspectors reviewed calibration records for ARMs and CAMs and interviewed health physics instrument technicians regarding the results. Whole Body Counter calibration records and daily source check trends were reviewed and discussed with a dosimetry technician and supervisor. The calibration records were inspected for a large sampling of the hand held instruments that were issued and in use.

Procedural guidance for the use and calibration of portable survey instruments was evaluated. The inspectors observed the daily source check records of survey meters and compared the results to specified tolerances. The inspectors interviewed an RP supervisor regarding the licensee's program for the use of electronic dosimeters (including those used in high noise areas) and observed workers using them at login/logout stations.

The licensee's respiratory protection program guidance and its implementation for SCBA use were evaluated and discussed with plant personnel. The number of available SCBA units and their general material and operating condition were observed during tours of the Control Room and Reactor Building and review of monthly inventory check list records. A physical operability inspection was performed on three pre-staged SCBA sets. The availability of different mask sizes for the pre-staged SCBAs was reviewed.

Program guidance, performance activities, and equipment material condition were reviewed against details documented in 10 CFR Parts 20 and 50; UFSAR Section 7.13, Area Monitoring; applicable sections of NUREG-0737, Clarification of Three Mile Island Action Plan Requirements, November 1980; RG 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Rev. 3 and RG 8.15, Acceptable Programs for Respiratory Protection, Rev. 1. Licensee procedures and activities related to SCBA were evaluated for consistency with TS and 10 CFR 20.1703. The licensee's instrumentation and protective equipment related procedures, reports and records reviewed during the inspection are listed in Section 2OS3 of the report Attachment.

Problem Identification and Resolution. Selected audits and self assessments associated with area radiation monitoring equipment, portable radiation detection instrumentation, and respiratory protective program activities were reviewed and assessed. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1.

The inspectors completed nine of nine required samples for IP 71121.03.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Effluent Monitoring and Radwaste Equipment. During inspector walk-downs, accessible sections of the Unit 1, Unit 2, and Unit 3 liquid and gaseous radioactive waste (radwaste) and effluent systems were assessed for material condition and conformance with system design diagrams. The inspection included floor drain tanks, reverse osmosis (Thermex) equipment, RHR Service Water Radiation Monitors (RM-90-133), Liquid Radwaste Effluent Monitor (RM-90-130), Turbine Building/Refuel Floor/Reactor Building/Radwaste Building Air Monitors (RM-90-250/252), Main Stack Monitors (RM-90-147/148), and associated airborne effluent sample lines. The inspectors interviewed chemistry supervision regarding radwaste equipment configuration requirements for representative sampling, and effluent monitor operation.

The inspectors reviewed performance records and calibration results for selected radiation monitors, flowmeters, and air filtration systems. For effluent monitors 2-RM-90-133D and 0-RM-90-147B/148B, the inspectors reviewed the last two loop/isotopic calibration records, the last two monthly source checks, and the last four quarterly functional tests. The last two flowmeter calibration records and High Efficiency Particulate Air (HEPA) surveillances for the Main Stack were also reviewed. The inspectors evaluated OOS effluent monitor logs and selected compensatory action data for the period February 2004 - December 2005.

Installed configuration, material condition, operability, and reliability of selected effluent sampling and monitoring equipment were reviewed against details documented in the following: 10 CFR Part 20; RG 1.21, Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials In Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants; American Nuclear Standards Institute N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities; TS Section 5; the Offsite Dose Calculation Manual (ODCM), Rev. 17; and UFSAR, Chapters 7 and 9. Procedures and records reviewed during the inspection are listed in Section 2PS1 of the report Attachment.

Effluent Release Processing and Quality Control (QC) Activities. The inspectors directly observed a release of liquid effluent that was monitored by 0-RM-90-130 and discussed release procedures with radwaste system operators. The inspectors also observed the preparation and counting of particulate, iodine, and tritium samples from weekly gaseous and liquid releases. Chemistry technician proficiency in collecting, processing, and counting the samples, as well as preparing the applicable release permits, was evaluated.

QC activities regarding gamma spectroscopy and beta-emitter detection were discussed with count room technicians and Chemistry supervision. The inspectors reviewed daily QC data logs from December 1, 2005 to January 31, 2006, for High Purity Germanium (HPGe) detectors No. 1 through 4 and reviewed licensee procedural guidance for count room QC. In addition, the inspectors reviewed the last two calibration records for HPGe detector No. 1. Results of the 2004 and 2005 radiochemistry cross-check program were also reviewed.

Selected parts of three procedures for effluent sampling, processing, and release were evaluated for consistency with licensee actions. One liquid and two gaseous release permits were reviewed against ODCM specifications for pre-release sampling and effluent monitor setpoints. The inspectors also reviewed the 2003 and 2004 annual effluent reports to evaluate reported doses to the public and to review ODCM changes.

Observed task evolutions, count room activities, and offsite dose results were evaluated against details and guidance documented in the following: 10 CFR Part 20 and Appendix I to 10 CFR Part 50; ODCM; RG 1.21; RG 1.33, Quality Assurance Program Requirements (Operation); and TS Section 5. Procedures and records reviewed during the inspection are listed in Section 2PS1 of the report Attachment.

Problem Identification and Resolution: Six PERs and one quality assurance audit associated with effluent release activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve selected issues in accordance with procedure SPP-3.1. Reviewed documents are listed in Section 2PS1 of the report Attachment.

The inspectors completed ten of ten required samples for IP 71122.01.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

REMP Implementation. Inspectors reviewed and discussed with cognizant licensee representatives the results published in the Browns Ferry Annual Radiological Environmental Operating report for CY 2003 and CY 2004. The inspectors observed the collection and preparation of weekly particulate and radio-iodine samples by licensee personnel and assessed material condition of seven air sampling stations (Station Nos. LM-1, LM-2, LM-3, LM-4, LM-6, LM-7, PM-3), one river water sampling station (Boat Dock Station), and five thermoluminescent dosimeters (Station Nos. 7, 9, 41, 44, 75) to evaluate procedural compliance. The inspectors assessed the calibration status of each air sampling pump. The inspectors also verified the placement of collection station locations against the sectors specified in the ODCM using an NRC global positioning system. The inspectors discussed with cognizant licensee representatives the procedures, methods, and equipment used to perform vegetation and sediment sampling. The inspectors reviewed and discussed with cognizant licensee representatives the procedures used to calibrate and determine the lower limit of detection for environmental sample gamma spectroscopy analysis.

REMP guidance, implementation, and results were reviewed against ODCM guidance and applicable procedures listed in Section 2PS3 of the report Attachment.

Meteorological Monitoring Program. The inspectors reviewed the operability of the meteorological monitoring equipment and operator access to meteorological data. Current meteorological monitoring equipment performance and calibration were reviewed. Licensee technicians primarily responsible for equipment maintenance and surveillance were interviewed by the inspectors concerning equipment performance, reliability, and routine inspections. Inspectors compared the meteorological data available in the control room against the meteorological data recorder at the tower location.

Meteorological instrument operation, calibration, and maintenance were reviewed against UFSAR, Chapter 2; NRC Safety Guide 23, Onsite Meteorological Programs-1972; and applicable licensee procedures. Documents reviewed are listed in section 2PS3 of the report Attachment.

Unrestricted Release of Materials from the Radiologically Controlled Area (RCA). Radiation protection activities associated with radioactive material control and the unconditional release of materials from the RCA were reviewed and evaluated. The inspectors observed surveys of personnel and material being released from the RCA and evaluated licensee response to detector alarms. The inspectors reviewed calibration records for two personnel contamination monitors and two material release monitors. Types of sources used for checks and minimum detectable activities were discussed with an instrument technician.

The inspectors verified that radiation detection sensitivities were consistent with NRC guidance in IE Circular 81-07 and IE Information Notice 85-92. Documents reviewed are listed in section 2PS3 of the report Attachment.

Problem Identification and Resolution. Licensee corrective action program (CAP) issues associated with environmental monitoring, meteorological monitoring, and release of materials were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1. Specific documents that were reviewed and evaluated in detail for these program areas are identified in Section 2PS3 of the report Attachment.

The inspectors completed ten of ten required samples for IP 71122.03.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Barrier Integrity Cornerstones

Reactor Coolant System Leakage and Reactor Coolant System Activity

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the following Performance Indicators (PI), including procedure SPP-3.4, Performance Indicator for NRC Reactor Oversight Process for Compiling and Reporting PI's to the NRC. The inspectors reviewed raw PI data for the PI's listed below for the first quarter 2004 through the fourth quarter 2005. The inspectors compared the licensee's raw data against graphical representations and specific values reported to the

NRC in the most recent PI report to verify that the data was correctly reflected in the report. The inspectors also reviewed the past history of PERs for any that might be relevant to problems with 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. Also, the inspectors witnessed the licensee's methods for actually collecting the PI data (i.e., RCS sample and analysis, and RCS leak measurement). The inspectors reviewed Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to verify that industry reporting guidelines were applied.

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

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety And Public Radiation Safety Cornerstones

a. Inspection Scope

The inspectors sampled licensee submittals for the PI indicated below for the period from January 2005 through December 2005. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 3, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed PER records generated from January 2005 through January 2006 to ensure that radiological occurrences were properly classified per NEI 99-02 guidance. The inspectors also reviewed electronic dosimeter alarm logs, radioactive material intake records, and monthly PI reports for calendar year 2005. In addition, licensee procedural guidance for classifying and reporting PI events was evaluated. Reviewed documents are listed in Section 4OA1 of the report Attachment.

Public Radiation Safety Cornerstone

- RETS/ODCM Radiological Effluents Occurrence

The inspectors reviewed records used by the licensee to identify occurrences of quarterly doses from liquid and gaseous effluents in excess of the values specified in NEI 99-02 guidance. Those records included monthly effluent dose calculations for calendar year 2005. The inspectors also interviewed licensee personnel that were responsible for collecting and reporting the PI data. In addition, licensee procedural guidance for classifying and reporting PI events was evaluated. Reviewed documents are listed in Section 4OA1 of the report Attachment.

The inspectors completed two of the required samples for IP 71151. One sample for the occupational radiation safety performance indicator and one sample for the public radiation safety performance indicator.

b. Findings

No findings of significance were identified.

4OA2 Identification & Resolution of Problems

.1 Routine Review of Problem Evaluation Reports

a. Inspection Scope

The inspectors performed a daily screening of all PERs entered into the licensee's corrective action program. The inspectors followed NRC Inspection Procedure 71152, "Identification and Resolution of Problems," in order to help identify repetitive equipment failures or specific human performance issues for follow-up.

b. Findings and Observations

There were no specific findings identified from this overall review of the PERs issued each day.

.2 Focused Annual Sample Review

The inspectors verified implementation of corrective actions associated with B Level PER 65268 related to an inadequate 10 CFR 50.59 evaluation and B Level PER 64835 related to a Unit 2 automatic scram due to a Turbine Generator Load Reject Signal.

a. Inspection Scope

The inspectors reviewed these two PER's, including related corrective action documents and root cause analysis, in detail to ensure that the full extent of the described issues were identified, a thorough evaluations were performed, and appropriate corrective actions were specified, prioritized, and completed. The inspectors also evaluated licensee actions against the requirements of the licensee's corrective action program as specified in SPP-3.1, Corrective Action Program, and 10 CFR 50, Appendix B.

b. Findings and Observations

PER 65268

Following a Unit 2 scram on July 8, 2004, the licensee implemented a temporary modification which removed the output of the power-load unbalance (PLU) circuit. Per licensee procedure, a 10 CFR 50.59 screening was performed to implement the temporary change. Following implementation and unit startup, it was determined that the 10 CFR 50.59 screening was inadequate and that a complete evaluation should have been performed. The licensee subsequently reinstated the PLU function. However, while the PLU function was removed, the licensee had been in violation of the Unit 2 TS (See LER 50-260/2004-003, and Inspection Reports (IR) 05000260/2005002 and 05000260/2004004).

The licensee determined that the root cause was inadequate information in the UFSAR. The licensee's corrective action plans included briefings for personnel responsible for 10 CFR 50.59 screenings and evaluations, additions to engineering periodic training, and extent of condition reviews. Corrective actions verified by the inspectors included changes to UFSAR Chapters 11 and 14 and briefing packages.

The change to UFSAR Chapter 11 added words that described the PLU circuit's function in a loss of load event. The changes in to Chapter 14, paragraph 14.5.2.1.1, Transient Description, stated specifically that "Turbine-generator power-load unbalance circuitry and other generator trips initiate turbine control valve (TCV) fast closure (minimum response time of TCV fast closure: 0.15 seconds)." This statement was determined to be misleading by the inspector because the PLU circuit is the only input to the fast acting solenoid valve portion of the hydraulic control system of the TCVs. It is the pressure decay of the oil system in the TCVs that causes the associated pressure switches to actuate input signals to the Reactor Protection System. The inspector discussed this issue with responsible engineering personnel who generated PER 100591 to clarify this statement.

PER 64835

While in steady state operation at 100% power, a Unit 2 main turbine trip/reactor scram occurred. All expected system responses were received, including the automatic opening of seven safety-relief valves. Electrical switching was in progress at the time of the scram, and during this switching activity the Unit 2 Unit Preferred System (UPS) 120 VAC Bus was inadvertently de-energized briefly. The reactor scram occurred at this time due to a turbine control valve fast closure/turbine trip condition. The loss of the UPS power would not by itself be expected to result in a turbine trip/reactor scram because of the fault-tolerant design of the main turbine electro-hydraulic control (EHC) system logic. However, it was determined that one of two main generator output current signal channels in the EHC logic had been automatically bypassed previously by the system software during a separate power supply transient on a different plant distribution bus.

The subsequent temporary interruption of the UPS bus caused the loss of the second main generator output current signal channel, and the system logic indicated that a power-load unbalance (i.e., main generator load reject) condition existed.

The licensee determined that the root cause was that the procedure controlling the transfer of the UPS bus contained inadequate detail to prevent interaction between the alternate and normal supplies' voltage control circuits. A contributing factor was an EHC system software configuration which gave the system an unrecognized single point scram vulnerability. (See LER 50-260/2004-001 and IR 05000260/2004004).

Implementation of corrective actions were verified by the inspector; these actions included revisions to applicable OIs and AOIs related to electrical alignments which could affect the EHC logic, changes implemented by WOs 04-718776-000 and 04-719089-000 for tuning of EHC logic parameters, and briefing packages for various groups of the plant staff.

4OA3 Event Follow-up

.1 (Closed) LER 05000260/2005-003-00, Reactor Protection System Actuation from Scram Discharge Volume High Level while Shutdown

On April 13, 2005, Unit 2 was in cold shutdown with surveillance testing being performed in support of returning the unit to power operations. At approximately 0954 hours, an unplanned actuation of one channel of the anticipated transient without scram/alternate rod insertion logic occurred. The actuation was caused by a subtle, unforeseen interaction between a surveillance test being actively performed and another test which had been temporarily halted for troubleshooting equipment problems. The plant equipment response to this logic actuation was to isolate and vent the CRD scram air header, causing the scram inlet and outlet valves on each CRD hydraulic control unit (HCU) to open and the scram discharge volume (SDV) vent and drain valves to close. With the scram outlet valves open, per the plant design a flow path existed from the reactor vessel, through the 185 individual control rod drives and the open scram outlet valve on each drive's HCU, to the associated SDV (east or west), causing each SDV to begin filling. Level switches on the associated scram discharge instrument volumes (SDIV) sensed the increasing water level, and at approximately 0955 hours Unit 2 received a reactor scram from high water level in both the east and west SDIVs. All control rods were fully inserted prior to the scram. There was no impact to the already shutdown unit as a result of the scram.

The LER and the associated PER 80721 were reviewed by the inspectors and no findings or performance deficiencies of significance were identified. This LER is closed.

.2 (Closed) LER 05000260/2005-005-00, Primary to Secondary Containment Leakage via the Residual Heat Removal System in Excess of Analyzed Limits

On September 2, 2004, with the RHR system in suppression pool cooling mode of operation, the control room received a high level annunciator alarm for the Pressure Suppression Chamber head tank. A leakage flow path was subsequently identified from the operating RHR system into this tank through two in-series check valves, which were part of the RHR system discharge piping keep-fill system. The primary containment isolation valve function of these leaking check valves was then declared inoperable, and the required TS actions were completed. The affected check valves were determined to be stuck open due to internal corrosion on the guide pin and disc, exacerbated by past cleaning practices and the use of dissimilar metals in the design.

This LER and the associated PER 85130 were reviewed by the inspectors. The enforcement aspects of this LER were previously addressed in IR 05000260/2005003, section 1R12.

Introduction: A Level IV NCV of 10 CFR 50.73(a)(2) was identified by the inspectors for the licensee's failure to submit an LER for a safety system functional failure of the Unit 2 residual heat removal pressure suppression chamber containment isolation valves.

Description: On September 2, 2004, the licensee discovered that both of the in-series keep-fill system check valves had stuck in the open position during standby service, such that they failed to check reverse system flow when the RHR system was started. With the check valves stuck in an open condition, had an actual design basis accident event occurred which involved significant fuel damage, utilization of this loop of RHR could have resulted in a release of highly radioactive water directly into the secondary containment beyond that analyzed in the BFN design and licensing bases.

However, this condition was not recognized as a reportable event pursuant to 10 CFR 50.73(a)(2)(v)(D) and (vii)(D) until identified by the inspectors. Consequently, the licensee did not submit an LER until September 19, 2005. Based on this, the inspectors concluded that the licensee failed to satisfy the reportability requirements of 10 CFR 50.73.

Analysis: As discussed in Section IV of the NRC Enforcement Policy, the significance of violations involving the failure to make required reports is not dispositioned using the Reactor Oversight Program's Significance Determination Process. The licensee's failure to provide a written event report does potentially impact the NRC's ability to carry out its regulatory function. However, because this failure to report per 10 CFR 50.73 did not actually impede or influence regulatory action, and the condition that required reporting under 10 CFR 50.73 was previously determined to be of very low safety significance (see IR 05000260/2005003), the NRC has characterized the significance of this reporting violation as a Severity Level IV in accordance with Section IV.A.3 and Supplement I of the NRC Enforcement Policy.

Enforcement: Pursuant to 10 CFR 50.73, the licensee shall submit an LER for any type of event described therein within 60 days after discovery of the event. Contrary to 10 CFR 50.73, the licensee failed to recognize and report within 60 days the aforementioned event which met the reporting requirements of 10 CFR 50.73(a)(2)(v)(D) and (vii)(D). Because the criteria of Section VI.a of the Enforcement Policy were satisfied, including the licensee's initiation of PER 99193, this violation will be considered an NCV: NCV 05000260/2006002-02, Failure To Report A Safety System Functional Failure Per 10 CFR 50.73. This LER is closed.

.3 <u>(Closed) LER 05000260/2005-007-00, Reactor Scram Due To Reactor Water Level Caused By Loss of Feedwater Pumps</u>

a. <u>Inspection Scope</u>

On August 5, 2005, Unit 2 automatically scrammed from 100% power due to low reactor vessel water level after a sequential loss of the 2C and then the 2B reactor feedwater pumps (RFP). The initial event followup was conducted per inspection procedure 71153 and documented in Section 4OA3.3 of IR 05000260/2005004. The inspectors have subsequently reviewed the LER and associated PERs 87178 and 87198, which included the root cause analysis and corrective action plans. The inspectors also interviewed responsible Operations and Maintenance department personnel. During the licensee's investigation, they confirmed that the initiating event was the unexpected loss of the 2C RFP. The licensee's root cause analysis also concluded that the root cause was a combination of poor workmanship when the 2C RFP main steam admission control valve linkage was reassembled, and procedural inadequacies which did not specify the required locking device or document its reinstallation. Furthermore, the inspectors verified that the corrective actions and extent of condition were consistent with the root cause(s). This LER is considered closed.

b. <u>Findings</u>

<u>Introduction</u>: A Green self-revealing finding was identified for inadequate work instructions and poor work practices associated with maintenance on the 2C reactor feedwater pump that resulted in a reactor trip.

<u>Description</u>: During the Unit 2 refueling outage in the Spring of 2005, maintenance was performed on the turbine-driven 2C RFP that necessitated disassembling the main steam flow control valve (FCV) linkage for the turbine. The 2C RFP was placed in service and Unit 2 was returned to full power on April 22, 2005. However, after approximately three months of operation, the 2C RFP main steam FCV linkage suddenly came apart causing the FCV to fail closed which resulted in a loss of motive force for the 2C RFP. Shortly after the 2C RFP ceased to function, the 2B RFP tripped due an apparently spurious actuation of the thrust bearing wear detector. Unit 2 was designed to accommodate the loss of one RFP at full power, without scramming, but not two. At this time, the NRC is not considering the failure of the 2B RFP to be the result of a licensee performance deficiency.

After the reactor scram, an inspection of the 2C RFP physical condition by the licensee discovered that the locking device (i.e., roll pin) had come loose from the FCV linkage allowing the fastener (i.e., threaded pivot pin with castellated nut) to become unfastened due to normal vibration. Additional inspection of the nut, pivot pin, and roll pin confirmed that the roll pin had been inadequately staked by the responsible mechanic(s). This specific activity was well within the expected skill-of-the-craft and considered to be poor workmanship. Furthermore, the licensee's review of the salient procedure, Mechanical Corrective Instruction (MCI) 0-003-TRB001, "Reactor Feedwater Pump Turbine And Turbine Components Disassembly, Inspection Rework And Reassembly," determined that this procedure was deficient. It did not provide any instructions regarding installation of a locking device (e.g., roll pin, split pin, or cotter pin) for the FCV linkage fastener where such a device was clearly needed. The responsible mechanic(s) had merely reinstalled the existing roll pin when they reassembled the linkage despite the lack of procedural guidance.

Analysis: This finding is greater than minor because it involved human error and inadequate work instructions that affected the human performance and procedure quality attributes of the Initiating Event Cornerstone to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because all safety-related mitigating systems operated as designed during and following the scram. The finding was entered into the licensee's corrective action program as PER 87178.

Enforcement: No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because it only involved non-safety related secondary plant equipment. This finding was of very low safety significance, and will be tracked as FIN 05000260/2006002-03, Poor Workmanship and Inadequate Work Instructions for Maintenance on the 2C Reactor Feedwater Pump That Resulted in a Reactor Scram.

4OA5 Other Activities

a. Inspection Scope

(Closed) Temporary Instruction (TI) 2515/161, Transport of Control Rod Drives in Type A Packages. The inspectors reviewed shipping logs and discussed shipment of CRDs in Type A packages with shipping staff. The inspectors noted that the licensee had conducted multiple shipments of CRDs in Type A packages since January 1, 2002. For these shipments, the inspectors reviewed and discussed Department of Transportation (DOT) requirements for proper Type A package use with responsible licensee shipping personnel. The inspectors reviewed licensee documentation of tests and engineering evaluations of packaging for compliance with DOT specifications. The inspection included a review of current vendor documentation for maintenance and package closure procedures.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

During the report period several interim inspection debriefs with senior management were conducted by regional inspectors. On April 6, 2006, the resident inspectors presented the integrated inspection results to Mr. John DeDomenico and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection period.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for disposition as an NCV.

- TS 5.5.4(c) requires the licensee to monitor effluents according to the methodology contained in the ODCM. ODCM Rev. 17, Section 1/2.1.2 states that any effluent monitor with an alarm/trip setpoint less conservative than the ODCM requires must be declared inoperable. Contrary to this, airborne effluent monitors RM-90-249/250/251/252 were allowed to operate with alarm setpoints for the noble gas channel that were less conservative than ODCM requirements. When the monitor were installed in 1991, the wrong units had been entered for flow rates in the applicable vents. The flows were entered in cc/sec rather than cc/min as required by vendor manuals, thus making the alarm setpoint a factor of 60 higher than desired (nonconservative). On May 3, 2004, an engineer reviewing the monitor vendor manuals discovered the error and took immediate corrective actions to correct the setpoints. This event is documented in the licensee's CAP as PER 60527. Although this event involved failure to maintain proper conservatism for noble gas effluent monitor alarm setpoints, this finding is of very low safety significance because the ability to assess dose to the public was not impaired, there were no releases above regulatory limits, and there are additional monitors that control automatic isolation of these airborne pathways.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Aukland, Nuclear Plant Manager
J. Burton Design Engineering Manager
J. Corey, Unit 1 Rad/Chem Manager
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
J. DeDimenico, Asst. Nuclear Plant Manager
R. DeLong, Site Engineering Manager
A. Elms, Nuclear Plant Operations Manager
A. Feltman, Emergency Preparedness Supervisor
A. Fletcher, Field Maintenance Superintendent
F. Froscello, ISI Supervisor
R. Jones, General Manager of Site Operations
F. Leonard, ISI Supervisor
L. Meyer, Site Nuclear Assurance Manager
R. Marks, Site Support Manager
R. Marsh, Operations Superintendent
D. Matherly, Nuclear Outage and Scheduling Manager
J. Mitchell, Site Security Manager
D. Nye, Maintenance & Modifications Manager
B. O'Grady, Site Vice President
C. Ottenfeld, Chemistry Manager
D. Sanchez, Training Manager
E. Scillian, Operations Training Manager
C. Sherman, Radiation Protection Manager
J. Sparks, Outage Manager
J. Steele, Outage Manager
K. Welch, Systems Engineering Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000260/2006002-01	NCV	Failure to Perform An Adequate Risk Assessment (1R13)
05000260/2006002-02	NCV	Failure To Report A Safety System Functional Failure Per 10 CFR 50.73 (Section 4OA3.2).
05000260/2006002-03	FIN	Poor Workmanship and Inadequate Work Instructions for Maintenance on the 2C Reactor Feedwater Pump That Resulted in a Reactor Scram (4OA3.3)

Closed

05000260/2005-003-00	LER	Reactor Protection System Actuation from Scram Discharge Volume High Level while Shutdown (4OA3.1)
05000260/2005-005-00	LER	Primary to Secondary Containment Leakage via the Residual Heat Removal System in Excess of Analyzed Limits (4OA3.2)
05000260/2005-007-00	LER	Reactor Scram Due To Reactor Water Level Caused By Loss of Feedwater Pumps (4OA3.3)
05000260/2515/161	TI	Transport of Control Rod Drives in Type A Packages (4OA5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluation of Changes, Tests, or Experiments

Full Evaluations

EDC 51557, Revise Drawing to Use Standard Disc in Lieu of V-notch Type, Rev. A
EDC 51631, Removal of Permali Shielding, Rev. A
TACF 3-04-004(5)(6)-003, U3 Reactor Feed Pump Woodward Emergency Shutdown Circuit
DCN 60561, Add Shutoff Valve to Reactor Head Vent Line, Rev. A
DCN 50939, Add Auto Changeover Bypass Switches, Rev. A
DCN T40676, Replace GE Magne Blast Breakers with Breakers which Use Vacuum Bottle Technology, Rev A. (Including PICs 50001A, 50031A, 50059A, 50121A, 50313A, 50427A, 50814A, 50994A, 63176A and FDCNs F40922A, F40949A, F41010, F41218A, F41422A)
DCN 51312, Replace Recirc. MG Sets 3A & 3B with Robicon Variable Frequency Drives (VFD), Rev. A
DCN 60536, Modify the Normal Alignment of the U3 RHR Cross-tie Loop by Closing BFN-3-SHV-74-150 and Opening BFN-3-FCV-74-46, Rev. A

Screened Out Items

DCN 62271, Provide Replacement for the 3C DG Air Compressor (Right Bank), Rev A
DCN 61488, Add Fuses in Panel 9-39 to Resolve HPCI Temperatures Above Essential Mild Conditions, Rev. A
DCN 64991, Provide an Additional Branch Connection on 2B RHRSW Supply Piping, Rev. A
DCN 51643, Modify Valve Internals to Allow for Improved Flow Characteristics, Rev. A
EDC 65424, Provide Material Equivalency Replacement for Hancock 5500 Valves on Feedwater Systems, Rev. A
DCN 60435 Modify Unit 3 EHC Control Software to Match Unit 2, Rev. A

EDC 64291, Valves to be Replaced with FAC Resistant Material i.e. Chrome-Moly, Rev A
DCN 62243, Reduce the Pressure Alarm Setpoint to a Value Lower than the Relief Valve Setting, Rev. A
DCN 60837, Replace Switches as Needed with New Model Switches, Rev. A
DCN 61310, Modify Unit 2 EHC Control Software to Match Unit 3, Rev. A
DCN 51295, Replace Hydraulic Snubber Supports with Mechanical Snubbers, Rev. B
DCN 63466, Provide Separate Power Supplies to U2 RFP Min. Flow Bypass Valves Control Circuits, Rev. A
DCN T26117A, Snubbers to be Replaced With Proper Flow Orientation
DCN 61985, Replace Pressure Indicators with Liquid-Filled Pressure Indicators, Rev. A
DCN 51070, Roll DC Solenoid Coil Wires and Replace Damaged Cable, Rev. A

Section 1R08: Inservice Inspection Activities
Problem Evaluation Report (PER) 87070, “OE 21150, Crack in FitzPatrick Torus”
PER 96089, “Ultrasonic Examination of RPV Nozzle N4D”
PER 98121, “3-CKV-001-0606"
PER 93949, “Unsatisfactory Final Liquid Penetrant Inspection on Stanchion Weld”
PER 84834, “OE 20902 Review”
PER 82784, “Generic Review-SQN PERs 80835 and 28319"
PER 88889, “Postponed ASME Sec XI Pressure Tests”
PER 94123, “Unit 3 CHECKWORKS Model Discrepancy”
PER 87900, “Pinhole Leaks - 2B/2D RHRSW Pipe Tunnel”
NDE Report R103, UT Examination in Weld GR-3-63, Unit 3 Cycle 10
U3C12-173, FAC Report for Component 3RCICZ-STMTRP-2E
U3C12-056, FAC Report for Component 3RFW1C-10E
Procedure N-UT-78, Generic Procedure for the Manual Ultrasonic Examination of Reactor Pressure Vessel Welds PDI-UT-6, Rev. 3
NDE Report R-182, UT Examination in Weld N4D-NV, Unit 3 Cycle 11
Procedure 54-ISI-858-02, Automated Ultrasonic Examination of Core Shroud Assembly Welds, Rev. 02/07/06

Section 1R17: Permanent Plant Modifications
MDQ3086-2004-0035, Evaluate Requirements of Replacement Air Compressor for the Diesel Generator Starting Air System, Rev. 0
WDP-BFN-3-BOP-013 (IAW DEDP-11, Design Walkdown Controls), Air Compressor 3-CMP-086-0668C Data for Replacement by DCN 62271, Rev. 0
MPI-0-000-BLT-001, Belt Drive Maintenance, Rev. 35
Rubber Manufacturers Association (RMA) IP-20-1988, Specifications for Drives Using Classical V-belts and Sheaves
PER 01-011604-000, Two WYLW/Siemens 4kV Vertical Lift Breakers Tripped Free During PMT After Installation
PER 03-003312-000, While Performing WO 01-009871-000 to Inspect 4kV SD BD “A”, the Breaker in Compartment 4 (Bkr 1824 Tie Bkr to 4kV SD BD “3EA”) Was Extremely Difficult to Rack Out of its Compartment
PER 03-003574-000, While Performing WO 01-009883-000 to Inspect 4kV SD BD “B”, the Breaker in Compartment 11 (“B” Electric Fire Pump) was Found With Damaged Ground Clip (Similar to PER 03-003312-000)

WOs 98-014890-001(079 and 099), Implement DCN T40676A, Stage 2 (30 and 57) to Replace the 4kV GE Magne Blast Breaker with Siemens Vacuum Type Breaker
EPI-0-000-BKR014, Inspection, Test, Checkout and Alignment of 4160 Volt Siemens (Type 3AF) Vacuum Circuit Breakers, Rev. 0013

Section 2OS1: Access Controls to Radiologically Significant Areas

Problem Evaluation Reports (PERs)
PER 73009, Emergent High Radiation Area Trend, 12/02/04
PER 81197, Contract RPT was briefed on correct RWP but signed onto the wrong one and received dose rate alarm while working in Moisture Separator Room, 04/2205
PER 81813, Two Maintenance Workers and an RPT were briefed on the correct RWP but were informed to sign in on the wrong RWP for work in the 2A RHR heat exchanger room, 05/02/05
PER 82025, AUO was briefed on correct RWP but signed onto the wrong one and received a dose rate alarm with no dose limit exceeded, 05/05/05
PER 87989, SRO going to relieve the on station SRO during the transfer of the MPC from the HTRAC to the STORM inadvertently crossed the HRA boundary without being on the HRA RWP but did not receive any dose, Was immediately stopped by HPTs controlling area, 08/21/05
PER 94017, Contract carpenter was briefed on correct RWP but signed onto the wrong one and received a dose rate alarm while working on scaffolding for U2 RHR activities, 12/14/05
PER 98679, Employee was briefed on the correct RWP but signed onto the wrong one and received a dose rate alarm while working in the Condensate Phase Separator Room in Radwaste, 03/08/06
PER 99003, A four wheeled cart that had been taken offsite was found to be contaminated, 03/11/06

Procedures
General Operating Instruction, 0-GOI-100-3B, Operations in Spent Fuel Storage Pool Only, Revision (Rev.) 33
RCI (Radiological Control Instructions) - 2.1, External Dosimetry Program Implementation, Rev. 52
RCI - 9.1, Radiation Work Permit Preparation and Administration, Rev. 45
RCI - 17, Control of High Radiation Areas and Very High Radiation Areas, Rev. 48
RCI - 23, Hot Spot Tracking Program, Rev. 7
RCI - 26, Radiation Protection Standards and Expectations, Rev. 3
SPP (Standard Programs and Processes) - 3.1, Corrective Action Program, Rev. 10
SPP - 5.1, Radiological Controls, Rev. 5

Radiation Work Permits (RWP)
RWP Number (No.) 06372153, U3C12 RXB RWCU HX Weld Repair (LHRA, Dose Control & Various Dress)
RWP No. 06372153, U3C12 RXB RWCU HX Weld Repair (Dose Control)
RWP No. 06372155, U3C12 RXB RWCU HX Weld Repair (LHRA, Respirator & Various Dress)
RWP No. 06372156, U3C12 RXB RWCU HX Weld Repair (LHRA, Respirator, Extremities & Various Dress)

Miscellaneous Documents

Browns Ferry Radiological Surveys, Nos. 030106-12 and 20 for M0203 Unit 3 (U3) Drywell 579'; 030206-21, M0026 U3 Drywell 584'; 030406-12, M0044 U3 Unit 3 RXB 593' RWCU Heat Exchanger Room (HER); 030606-8 and 26 for M0044 U3 RXB 593' RWCU HER; 030706-11 and 40 for M0050 U3 Drywell 616'; 030706-18, M0044 U3 RXB 593' RWCU HER; 030906-8 and 28 for M0026 U3 Drywell 584'; 031106-31, M0044 U3 RXB 593' RWCU HER; 031306-9, M0044 U3 RXB 593' RWCU HER; and 031406-6 and 20 for M0044 U3 RXB 593' RWCU HER
NRC 06 High Radiation Trending Document
NRC 06 RWP Compliance Report
Radiation Protection Integrated Analysis, First Half Fiscal Year (FY) 2005
Radiation Protection Integrated Trend Reviews, Third Quarter (QTR) FY 2005, Fourth QTR FY 2005, and First QTR FY 2006
RCI-17, Rev. 0048, Attachment 4, Locked High Radiation Areas, Dated 03/14/06

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

Reports, Procedures, Instructions, Lesson Plans and Manuals

Browns Ferry Nuclear Plant FSAR Section 7.13 (Area Radiation Monitoring System), Amendment 21
RCI-1.1 Part 1, Field Operations Program Implementation, Rev. 116
RCI-1.1 Part 2, Determination of Respiratory Protection Requirements, Rev. 0116
RCI-3.1, Respiratory Protection Program Implementation, Rev. 27 and 28
RCI-11.1 Parts 1-3, Radiation Protection Instrument Program Implementation, Rev. 64
RCDP-1, Conduct of Radiological Controls, Rev. 2
SPP- 5.1, Radiological Controls (Section 3.9), Rev.5
(Surveillance Instruction) 3-SI-4.2.K.2.d, Reactor Building Vent Exhaust Monitor Sample Flow Calibration and Functional Test 3-RM-90-250, Rev. 13
3-SI-4.2.K.2.a, Reactor Building Vent Exhaust Monitor Sample Flow Calibration and Functional Test 3-RM-90-250, Rev. 17, 11/22/04
CI-303.13, Energy Calibration and Daily Checks (Gamma Spectroscopy System), Rev. 9
CI-303.15, Efficiency Calibration (Gamma-ray Spectrometry System), Rev. 13
CI-703, Sample Preparation for Gamma Ray Spectroscopy, Rev. 18
SCBA Maintenance Records for 5 year maintenance to include testing for 30 SCBA units, CY2005
MSA Training Certificate for 6 personnel authorized to perform repairs on MSA SCBAs with Belt Mounted Regulators, 3/4/2005
MSA Training Certificate for 5 personnel authorized to perform repairs on MSA SCBAs with Mask Mounted Regulators, 8/4/2005
Calibration Data Sheet, Eberline RO-20 SN 4824, 1/17/2006
Calibration Data Sheet, Eberline RO-20 SN 5645, 12/5/2005
Calibration Data Sheet, Eberline RO-20 SN 4829, 12/13/2005
Calibration Data Sheet, Bicron RSO-5 SN A916G, 12/12/2005
Calibration Data Sheet, Bicron RSO-50, SN A809D, 10/19/2005
Calibration Data Sheet, Bicron RSO-50, SN A983B, 12/6/2005
Calibration Data Sheet, Bicron RSO-50 SN A113E, 12/13/2005

CAP Documents

Browns Ferry Nuclear Plant (BFN) - Nuclear Assurance (NA) -Oversight Report for the Period of November 21, 2003 Through January 20, 2004 - NA-BF-04-004

BFN-NA -Oversight Report for the Period of January 21, 2004 Through March 20, 2004 - NA-BF-04-007

BFN-NA- Oversight Report for the Period of March 21, 2004 Through May 20, 2004 - NA-BF-04-008

BFN-NA- Oversight Report for the Period of May 21, 2004 Through August 20, 2004 - NA-BF-04-012

BFN- NA- Oversight Report for the Period of January 1, 2005 Through April 19, 2005 - NA-BF-05-019

BFN- NA- Oversight Report for the Period of April 20, 2005 Through June 30, 2005 - NA-BF-05-022

BFN- NA- Oversight Report for the Period of July 1, 2005 Through September 30, 2005 - NA-BF-05-034

Nuclear Assurance (NA)-TVAN Wide- Audit Report No. SSA0302- Radiological Protection and Control Audit, December 31, 2003

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

Procedures

0-TI-45, Liquid Process Radiation Monitors, Rev. 14

0-TI-15, Radioactive Gaseous Effluent Engineering Calculations and Measurements, Rev. 13

CI-703, Sample Preparation for Gamma Ray Spectroscopy, Rev. 18

CI-724, Preparation of Samples for Liquid Scintillation Analysis, Rev. 11

0-SI-4.8.B.2-1, Airborne Effluent Analysis - Particulate and Charcoal Filter Analysis, Rev. 30

CI-303.13, Energy Calibration and Daily Checks (Gamma Spectroscopy System), Rev. 9

SPP-3.1, Corrective Action Program, Rev. 10

Records and Data

Annual Radioactive Effluent Release Report, 2003 and 2004

Radiation Monitor 2-RM-90-133D Calibrations, 12/22/03 and 11/18/04

Radiation Monitors 0-RM-90-147B/148B Calibrations, 2/8/04 and 4/23/05

Radiation Monitor 2-RM-90-133D Monthly Source Checks, 12/15/05 and 1/11/06

Radiation Monitors 0-RM-90-147B/148B Monthly Source Checks, 12/6/05 and 1/3/06

Radiation Monitor 2-RM-90-133D Quarterly Functional Tests, 4/14/05, 5/20/05, 8/4/05, 1/17/06 Radiation Monitors 0-RM-90-147B/148B Quarterly Functional Tests, 1/3/05, 6/23/05, 9/16/05, 12/8/05

Main Stack Flowmeter Calibrations, 4/23/05 and 7/27/05

Germanium Detector No. 1 Calibrations, various geometries, 2002 and 2003

Gaseous Radioactive Waste Release Permit Nos. 60059.030.004.G and 60060.031.004.G

Liquid Radioactive Waste Release Permit No. 60048.004.013.L (½6/06)

Results of Radiochemistry Cross-Check Program, 2004 and 2005

Germanium Detectors No. 1 - 4, Daily Source Check and Trend Logs, 12/1/05 - 1/31/06

Main Stack HEPA Performance Quarterly Test, 10/13/05 and 1/6/06

Out-of-service data for radiation monitors 0-RM-90-147/148 and 2-RM-90-133D, February 2004 - December 2005

CAP Documents
Nuclear Assurance Audit No. SSA0502, Radiological Protection and Control Audit, 1/19/06
PER 60527, Non-conservative alarm setpoints discovered on airborne effluent monitors, 5/3/04
PER 79075, Contamination found in auxiliary decay heat removal system, 3/21/05
PER 81265, Liquid radwaste release duration > 30 days with inoperable flow rate instrument, 4/25/05
PER 79143, Contamination found in residual heat removal system, 3/21/05
PER 89288, Liquid radwaste release duration > 30 days with inoperable flow rate instrument, 9/14/05
PER 96105, Tritium contamination found in onsite groundwater wells, ½6/06

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

Procedures, Manuals, and Guidance Documents

Browns Ferry Nuclear Plant Offsite Dose Calculation Manual (ODCM), Rev. 16 and 17
QA Plan, Quality Assurance Program Implementation for BFNP REMP, Rev. 11
SPP-3.1, Corrective Action Program, Rev. 10
EMSTD-01, Environmental Radiological Monitoring Program, Rev. 22
ENV-01, Preparation of Annual Radiological Environmental Operating Reports, Rev. 10
ENVR-04, Special Reports of Radiological Environmental Monitoring Program Results, Rev. 13
ENV-05, Land Use Surveys, Rev. 9
LSAP-0005, Training and Qualification of Radiochemical Laboratory Technicians, Rev. 12
Radiological Control Instruction (RCI)-1.1, Field Operations Implementing Procedure (FO-IP) No. 1, Personnel Decontamination, Rev. 0116
RCI-1.1, FO-IP No. 3, Alarm Response, Rev. 0116
RCI-1.1, FO-IP No. 4, Radionuclide Trending and Assessment Program, Rev. 0116
RCI-1.1, FO-IP No. 8, Standardized Radiological Postings, Rev. 0116
RCI-1.1, FO-IP No. 9, Radiation and Contamination Surveys, Rev. 0116
RCI-1.1, FO-IP No. 12, RP Environmental Monitoring Survey Program, Rev. 0116
SC-03, Calibration Procedure for Radiological Environmental Monitoring Air Sampler System Gas Meter, Rev. 4
Radiological Environmental Monitoring - Atmospheric Air Sampler Baseline Description of System Configuration, Approved 09/18/1997
Meteorological Monitoring, Baseline Description of System Configuration, Rev. 16

Records, Data, and Annual Reports

Browns Ferry Nuclear Plant 2003 Annual Radiological Environmental Operating Report
Browns Ferry Nuclear Plant 2004 Annual Radiological Environmental Operating Report
Browns Ferry Meteorological Data Recoverability Reports for January - March 2005, April - June 2005, and July - September 2005
Certification/Qualification Records for E. Smith, dated 12/01/04 and 12/02/04
Results of Environmental External Cross Check Program for 2003, 2004, and 2005
Calibration Data Sheets for REMP Air Sampler Gas Meters: LM-1, 1/09/06; LM-2, 1/09/06; LM-3, 1/09/06; LM-4A, 1/09/06; LM-4B, 1/09/06; LM-6, 1/09/06; LM-7, 1/09/06;

PM-1A, 1/09/06; PM-1B, 1/09/06; PM-2A, 1/09/06; PM-2B, 1/09/06; PM-3, 1/09/06; and RM-1, 1/09/06
Sonic Wind Direction Calibration Sheet for 10 m, 46 m, and 91 m, 04/18/05 and 10/24/05
Sonic Wind Speed Calibration Sheet for 10 m, 46 m, and 91 m, 04/18/05 and 10/24/05
Air Temperature System Calibration Sheet for 10 m, 46 m, and 91 m, 04/18/05 and 10/24/05
10 CFR 61 Analysis Report for DAW smears (Unit 0, 2, & 3), 09/28/05
SAM-11Calibration Form: TVA Tag No. 860067, 12/04/05; TVA Tag No. 860252, 10/07/05
Eberline Instrument Corporation PCM-2 Calibration Form: TVA Tag No. 592, 12/10/05; TVA Tag No. 594, 12/10/05

CAP Program Documents

Self-Assessment CRP-BPS-05-002, Radiological Environmental Monitoring Program, 03/25/2005
Self-Assessment CRP-ERMI-04-001, Effectiveness of Environmental Radiological Monitoring and Instrumentation (ERM&I) Laboratory Quality Control (QC) Program, 03/25/2005
Nuclear Assurance (NA) Audit Report No. SSA0302 - Radiological Protection and Control Audit, 12/31/2003
PER 60169, BFN REMP missed samples, 4/28/04
PER 70931, Missed REMP samples BFN RM-1, 10/27/04
PER 75851, BFN REMP missed sample, 2/01/05
PER 81888, REMP missed samples BFN, 5/03/05
PER 94343, Missed REMP samples SQN, 12/21/05

4OA1 Performance Indicator Verification

Occupational and Public Radiation Safety Cornerstones

Procedures and Records

Semi-Annual Radioactive Effluent Release Report, Doses to a Member of the Public Due to Liquid Releases, January - December 2005
Semi- Annual Radioactive Effluent Release Report Doses to a Member of the Public Due to Radio-iodines, Tritium, and Particulates in Gaseous Releases, January - December 2005
Occupational Radiation Safety Event Reporting Spreadsheet Tracking number of TS high radiation area occurrences, number of very high radiation area occurrences , and number of unintended exposure occurrences November 2003 through January 2006
Radiation Protection Integrated Trend Review -First Half 2005
Radiation Protection Integrated Trend Review -3rd Quarter 2005
Radiation Protection Integrated Trend Review 4th Quarter 2005

CAP Documents
PER 69587, Workers not briefed on radiological conditions in equipment pit.
PER 81197, A Radiation Protection Technician failed to change his RWP entry to the proper Locked High Radiation Area RWP and received a dose alarm with 85 mrem accrued vs. a dose alarm setpoint of 80 mrem.

PER 81813, Workers entered a Locked High Radiation Area on the wrong RWP. The workers were briefed on the correct RWP and conditions but were inadvertently informed to use the wrong RWP.
PER 82194, A worker received a dose rate alarm while working on a drain line on 2 A RHR HX.
PER 87213, On 8/6/05 an employee told RP the incorrect work location for his assigned work. The worker stated that he would be working in the HPCI area when in fact he was working in the RCIC area.

4OA5 TI 2515/161, Transport of Control Rod Drives (CRD) in Type A Packages

TVA Browns Ferry Shipping Logs for shipments from January 1, 2002 to January 3, 2006
Shipping Paperwork for Shipment Number 030311, 11/14/2005 (CRD Shipment)
Shipping Paperwork for Shipment Number 050401, 11/14/2005 (CRD Shipment)
Shipping Paperwork for Shipment Number 051121, 11/14/2005 (CRD Shipment)
PER 92265, Contaminated water leaked from CRD shipping box, 11/09/05
OE195351, Lid Bolt Torque Requirement Missed for Type A CRD Shipping, 11/18/04
SC-01-1247, Container Products Corporation (CPC) Package Certification Documentation, Rev. 0
SC-01-1524, CPC Packaging Certification Document, Rev. C
SPEC-01-1524, CPC Package Certification Documentation, Rev. 1
CHP-100, CPC Specifications for Package Handling, Maintenance, and Inspection

ACRONYMS

ADHR	auxiliary decay heat removal
AOI	abnormal operating instructions
ARM	area radiation monitor
ASME	American Society of Mechanical Engineers
BFN	Browns Ferry Nuclear
CAM	continuous air monitor
CAP	corrective action program
CRD	control rod drive
CS	core spray
DOT	Department of Transportation
EDG	emergency diesel generator
EECW	emergency equipment cooling water
EHC	electro-hydraulic control
EOI	emergency operating instructions
FA	fire area
FAC	flow assisted corrosion
FCV	flow control valve
FZ	fire zone
FIN	Finding
HCU	hydraulic control unit
HPCI	high pressure coolant injection
HPGe	high purity germanium
HPT	health physics technician
ISI	inservice inspection
IST	inservice testing
LHRA	locked high radiation area
MCI	Mechanical Corrective Instruction
MFW	main feedwater
MR	maintenance rule
MS	main steam
MT	magnetic particle testing
NCV	Non-Cited Violation
NDE	nondestructive examination
NEI	Nuclear Energy Institute
No.	number
ODCM	Offsite Dose Calculation Manual
OI	operating instructions
OOS	out-of-service
ORAM	Outage Risk Assessment Management
P&ID	piping and instrumentation drawing
PER	Problem Evaluation Report
PI	Performance Indicator
PLU	power-load unbalance
PMT	post-maintenance testing
PSA	probabilistic safety analysis
QC	quality control

radwaste	radioactive waste
RCA	radiologically controlled area
RCP	reactivity control plan
RCS	reactor coolant system
RE	reactor engineers
RFP	reactor feedwater pump
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RMA	risk management actions
RMOV	reactor motor-operated valve
ROP	Reactor Oversight Program
RP	radiation protection
RT	radiographic testing
RWP	radiation work permit
RWR	reactor water recirculation
SCBA	self-contained breathing apparatus
SDP	Significance Determination Process
SDV	scram discharge volume
SLC	standby liquid control
SPP	Standard Program and Process
TCV	turbine control valve
TS	Technical Specifications
U3C12	Unit 3 Cycle 12 Refueling Outage
UFSAR	Updated Final Safety Analysis Report
UPS	unit preferred system
VFD	variable frequency drive
VT	visual testing
WO	work order