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

April 30, 2007

Tennessee Valley Authority
ATTN: Mr. Preston D. Swafford
Interim Chief Nuclear Officer
and Senior Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2007002, 05000260/2007002, AND 05000296/2007002 AND ANNUAL ASSESSMENT MEETING SUMMARY

Dear Mr. Swafford:

On March 31, 2007, 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 5, 2007, with Mr. Brian O'Grady 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. The principal 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 three NRC-identified findings and two self-revealing findings of very low safety significance (Green), which were determined to be violations of NRC requirements. However, because these findings were of very low safety significance and were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCV) consistent with Section VI.A of the NRC Enforcement Policy. In addition, two licensee-identified violations, which were determined to be of very low safety significance, are listed in Section 4OA7 of this report. If you contest any non-cited violation 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 05000259/2006005, 05000260/2006005, and 05000296/2006005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at the Browns Ferry Nuclear Plant.

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Letter to Preston D. Swafford from Malcolm T. Widmann dated April 30, 2007
SUBJECT: BROWNS FERRY NUCLEAR PLANT - INTEGRATED INSPECTION REPORT 05000259/2007002, 05000260/2007002, and 05000296/2007002

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REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report Nos.:	05000259/2007002, 05000260/2007002, and 05000296/2007002
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, 2007
Inspectors:	T. Ross, Senior Resident Inspector R. Monk, Resident Inspector C. Stancil, Resident Inspector J. Griffis, Health Physicist (Sections 2OS1, 2PS1, and 4OA1.2) R. Hamilton, Senior Health Physicist (Sections 2OS2 and 4OA1) R. Lewis, Reactor Engineer (Sections 1R04 and 1R12) W. Loo, Senior Health Physicist (Section 2PS2) E. Michel, Reactor Inspector (Section 1R08)
Approved by:	Malcolm T. Widmann, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

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(Note: Page numbers above may not exactly match the report depending on the printer used.)

SUMMARY OF FINDINGS

IR 05000259/2007-002, 05000260/2007-002, 05000296/2007-002; 01/01/2007 - 03/31/2007; Browns Ferry Nuclear Plant, Units 1, 2, and 3; Refueling and Other Outage Activities, Radioactive Material Processing and Transportation, Access Control To Radiologically Significant Areas, Identification and Resolution of Problems, and Other.

The report covered a three-month period of routine inspections by the resident inspectors and the following regional personnel: two senior health physicists, a health physicist and two reactor engineers. Five Green findings, all of which were non-cited violations (NCVs), were identified. The significance of most findings are 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green non-cited violation of 10CFR50, Appendix B, Criterion XVI, for ineffective corrective actions by the licensee to ensure that the operating emergency diesel generators (EDGs) during a Unit 2 Station Blackout (SBO) event would have sufficient cooling water under worst case licensing-basis conditions. The licensee developed a new simplified mitigation strategy to address the issue. This finding was entered into the licensee's corrective action program as PER 119778.

  This finding was greater than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance because of the low frequency of occurrence of the specific combination of multiple EDG failures that could lead to a loss of cooling water flow to all of the running EDGs. The cause of finding was directly related to the appropriate and timely corrective action aspect of the Problem Identification and Resolution cross-cutting area because corrective actions developed for Unit 2 SBO mitigation strategy deficiencies were not effective in ensuring timely restoration of cooling water to the EDGs. (Section 4OA2.2)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green non-cited violation of TS 5.4.1.a for the operators failure to maintain Unit 2 core flow within the bounds of the Core Power/Flow Map established by operating procedures. When notified, the licensee promptly reduced reactor recirculation pump flow. This finding was entered into the licensee's corrective action program as PER 119305.

This finding was more than minor because if left uncorrected, operators could have unknowingly allowed core flow to exceed the analytical bounds of the fuel vendor's reload report transient analysis which would have been a more significant safety concern. This finding was of very low safety significance because it is associated with fuel barrier integrity. Furthermore, core flow was still within the envelope of the fuel vendor's analytical limits and none of the reactor fuel thermal limits were exceeded. The cause of the finding was directly related to the procedure compliance aspects of the Human Performance cross-cutting area because of inadequate communication of management and supervisory expectations for unit operations in the increased core flow region and lack of operator attention to the proceduralized power/flow map limits. (Section 4OA5.3)

Cornerstone: Occupational Radiation Safety

- Green. Two examples of a Green self-revealing non-cited violation of 10 CFR 20.1501(a)(2)(i) were identified for failure to conduct surveys that were reasonable under the circumstances to evaluate the magnitude and extent of radiation levels in areas where work was performed. On February 9 and 16, 2007, investigation into electronic dosimeter dose rate alarms, received during work activities, revealed dose rates in excess of those measured during pre-job surveys. Since the new dose rates exceeded the criteria for posting as a high radiation area, the licensee immediately posted and controlled these areas as high radiation areas. This finding was entered into the licensee's corrective action program as PERs 119482 and 119829.

  This finding is more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process, and it adversely affected the cornerstone objective because the failure to conduct adequate surveys did not ensure adequate protection of worker health and safety from exposure to radiation. Using the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because the failures to survey did not pose a substantial potential for over exposure and did not affect the ability to assess doses. The cause of the finding was directly related to the work activity coordination aspect of the human performance cross-cutting area because pre-job radiological surveys were inadequate to apprise personnel of plant conditions that affected work activities. (Section 2OS1)

Cornerstone: Public Radiation Safety

- Green. A Green self-revealing non-cited violation of 10 CFR 71.5 was identified for failure to properly package radiological material such that, under conditions normally incident to transportation, the radiation levels at the external surface of the package would not exceed applicable Department of Transportation (DOT) limits. When the two shipments arrived at a processing facility on April 21, 2005, the radiation dose rates measured on portions of the external surface of the packages were as high as 300 mrem/hr, which was in excess of the 200 mrem/hr limit specified by the regulation. The licensee established additional supervisory review and approval prior to shipping packages approaching DOT limits. This finding was entered into the licensee's corrective action program as PER 81364.

This finding is more than minor because it is associated with the Plant Facilities/ Equipment and Instrument attribute of the Public Radiation Safety cornerstone and adversely affected the cornerstone objective, in that, the improper transportation packaging resulted in a shipping container with external dose levels exceeding regulatory requirements. Using the Public Radiation Significance Determination Process, the finding was determined to be of very low safety significance because the areas on the packages with elevated radiation levels were inaccessible to the public and the radiation levels were less than two times the DOT limit. (Section 2PS2)

Miscellaneous

- Green. The inspectors identified a Green non-cited violation of Technical Specification 5.2.2.d due to inadequate management oversight and awareness of the administrative requirements for controlling overtime which resulted in multiple instances of Instrumentation and Control personnel exceeding overtime limits without prior authorization and documentation. Management immediately changed work schedules to comply with the Technical Specification requirements and entered the issue into their corrective action program as PER 119016.

  This finding was greater than minor because if left uncorrected it could become a more significant safety concern due to excessive fatigue by key maintenance personnel performing safety-related activities. An NRC management review determined that the finding was of very low safety significance because no specific performance deficiencies were identified for the individuals during the time they exceeded the established overtime limits. (Section 4OA5.2)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 began the report period in Mode 5 with the core fully loaded. The unit entered Mode 4 (i.e., reactor vessel head fully tensioned) on February 28, 2007, and remained in Mode 4 for the rest of the report period.

Unit 2 began the report period at full power and operated at full power until an automatic reactor scram occurred on January 11, 2007. The unit was restarted the next day and returned to full power by January 16. Shortly thereafter, Unit 2 began an extended end-of-cycle (EOC) coastdown which continued until it was shutdown on February 20 for the Unit 2 Cycle 14 (U2C14) refueling outage. By the end of the report period, Unit 2 was still in Mode 5 for the U2C14 outage.

Unit 3 operated at essentially full power for the entire report period, except for an automatic reactor scram on February 9, and an unplanned downpower to 80% on February 17 to replace a 3B Condensate Booster pump seal.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On March 1, a Tornado Watch was declared for adjacent counties. The inspectors observed the licensee's implementation of abnormal operating instruction (AOI)100-7, Tornado. The inspectors also reviewed and discussed the implementation of AOI 100-7 with the responsible Unit Supervisor (US) and Shift Manager. Furthermore, the inspectors witnessed the licensee's execution of evacuation orders of vulnerable areas and buildings outside the power block, and the termination of work and evacuation of the turbine and refueling floors.

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, 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 2 High Pressure Coolant Injection (HPCI) system
- Unit 2 Reactor Core Isolation Cooling (RCIC) system
- Unit 3 Division I Residual Heat Removal (RHR) system

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 1 and 2 B Emergency Diesel Generator (EDG), using the applicable P&ID flow diagrams, 0-47E861-2A, 0-47E861-5, and 0-47E840-3, along with the relevant operating instructions, 0-OI-18 and 0-OI-82, to verify equipment availability and operability. The inspectors reviewed relevant portions of the Updated Final Safety Analysis Report (UFSAR) and TS. This detailed walkdown also verified electrical power alignment, the condition of applicable system instrumentation and controls, component labeling, pipe hangers and support installation, and associated support systems status. Furthermore, the inspectors examined the applicable System Health Reports, Work Orders, and any Problem Evaluation Reports (PERs) that could affect system alignment and operability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Walkdowns

a. Inspection Scope

Walkdowns. The inspectors reviewed licensee procedures, Standard Programs and Processes (SPP)-10.10, Control of Transient Combustibles, and SPP-10.9, Control of Fire Protection Impairments, and conducted a walkdown of the nine 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 features 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.

- Fire Area 8 (4Kv Shutdown Board Room D)
- Fire Area 25 (Intake Structure and RHR Service Water (RHRSW) Pump Rooms)
- Fire Area 23 (4Kv Shutdown Board Room 3EC and 3ED)
- Fire Area 13 (4Kv Shutdown Board Room E)
- Fire Area 21 (Unit 3 EDG Building)
- Fire Area 22 (4kv Shutdown Board Room 3EA and 3EB)
- Fire Area 19 (Unit 3 Battery Room and Battery Board Room)
- Fire Area 4 (Unit 1B Electric Board Room)
- Fire Zone 2-1(Unit 2 Reactor Building Elevation 565)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

Piping Systems ISI

a. Inspection Scope

The inspectors observed and reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries for Brown's Ferry Unit 2. The inspectors selected a sample of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI required examinations and non-Section XI inspections of other code components in order of risk priority as identified in Section 03 of inspection procedure 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the onsite inspection period. The inspectors also reviewed a sample of inspection activities associated with components that are outside the scope of ASME Section XI requirements, which are performed in accordance with commitments to follow industry guidance documents, such as the Boiling Water Reactor Vessel and Internals Project (BWRVIP). The inspectors reviewed the Outage Summary Report (NIS-1), dated July 15, 2005, for the last Unit 2 refueling outage.

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate the licensee's compliance with Technical Specifications (TS); ASME Section XI, and Section V, 1995 Edition through the 1996 Addenda for Class 1, 2, and 3 systems; and BWRVIP documents for the inspection of Reactor Vessel Internals. For Brown's Ferry Unit 2, this was the second outage of the second period of the third interval. The inspectors verified that indications and defects were appropriately evaluated and dispositioned in accordance with the applicable requirements of the ASME Section XI Code, and the BWRVIP documents. Specifically, the inspectors observed the following examinations:

Manual Ultrasonic Testing (UT)
- DRHR-2-12, Residual Heat Removal, Flued Pipe to Valve (2-FCV-074-67), 24"

The inspectors also observed the conduct of BWR In Vessel Visual Examinations (IVVI) of jet pump retaining wedges, and UT of shroud horizontal welds in accordance with the BWRVIP. Specifically, the inspectors reviewed the following examination records:

Ultrasonic Testing (UT)
- RHRG-2-09-A, Core Spray Heat Exchanger Shell weld, 54"

Radiographic Testing (RT)
- HPCI-2-001-004, pup-pipe shop weld for replacement of 2-FCV-073-002, 10"

Liquid Penetrant Exam (PT)
- HPCI-2-001-004, pup pipe shop weld for replacement of 2-FCV-073-002, 10"

The inspectors also reviewed the following examination records that contained recordable indications:

- DW LNR-2-1, Drywell liner moisture seal
- PEN 2-X-5G, Main Vent Line and Vent Header

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 requirements stated in ASME Section V, ASME Section XI, BWRVIP documents, and other industry standards.

The inspectors reviewed welding activities associated with WO 00-003350-00, and Design Change Package (DCN) 50287. The associated ASME Class 1 pipe and valve were welded together in the shop as part of a replacement project for valve 2-FCV-073-002. The inspectors reviewed the welding procedures, applicable procedure qualification records, welder performance qualification test records, and material certifications for compliance to ASME Section IX requirements. The following welds were reviewed by the inspectors:

- HPCI-2-001-004, Pipe to valve weld, 10"

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

The inspectors discussed and reviewed the licensee's preliminary response to the 10CFR50.73(b)(3)(ii) report dated February 19, 2007, regarding substantial cracking in dissimilar metal welds in BWRs. The licensee's preliminary response was in compliance with the BWRVIP-75-A.

Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On February 5, 2007, the inspectors observed the as-found simulator evaluations for two crews per OPL178.063, "NI Failure, Loss of 3D EDG, LOSP, HPCI Failure, LOCA, Diesel and Core Spray Failures." The Loss of Offsite Power with diesel failures led to a Notice of Unusual Event declaration; followed by an Alert declaration for the Loss of Coolant Accident (i.e., Drywell pressure >2.45 psig).

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)
- 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 inspectors also attended the 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

a. Inspection Scope

The inspectors reviewed the two systems listed below 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). Both of these systems had exceeded their reliability performance criteria and were classified as (a)(1). 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, surveillance records, PERs, system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met.

- Hydrogen/0xygen Analyzers
- Unit Preferred Motor Generator Sets

b. Findings

No findings of significance were identified

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For the six planned online work and/or emergent work, that affected the 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.

- Unit 3 RCIC & Unit 3A CRD OOS
- Unit 2 RCIC and Unit 2A RHR Heat Exchanger OOS
- Unit 2 HPCI, 2D RHR Heat Exchanger and B1 RHRSW Pump OOS
- Unit 2 Operations with Potential for Draining the Reactor Vessel (OPDRV), such as Local Power Range Monitor detector replacement and 2B Recirculation pump/motor replacement
- Unit 3 activities for work week 2711 during refueling outage U2C14
- Secondary Containment damper replacements, D EDG OOS and Standby Coolant to Unit 3 OOS

b. Findings

No findings of significance were identified

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the seven operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedure 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 on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- 1A Shutdown Board Room Fire Barrier Seals (PER 115787)
- Unit 3 RCIC Pump (PER 119628)
- Unit 2 Station Blackout (SBO) with Multiple EDG Failures (PER 114913 and 114967)
- RHRSW Heat Exchanger Inlet Check Valves (PER 116511)
- North Header Emergency Equipment Cooling Water (EECW) check valve 0-CKV-67-671 (PER 121901)
- Unit 2 Operation Outside Core Power-Flow Map (PER 119305)
- Unit 2 Recirculation Loop B Jet Pump Leakage (PER 120729)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the Design Change Notice (DCN) and completed work package for DCN 63471A, Modify 3A Electric Board Room Air Conditioner to allow it to run continuously under low load conditions, including related documents and procedures. The inspectors also observed the following activities: portions of work associated with WO 05-723431-000 and WO 05-723432-000; field installation; and post modification testing.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance tests (PMT) listed below to verify that procedures and test activities confirmed system, structure, or component (SSC) operability and functional capability following maintenance. The inspectors reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested; that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents; and, that the procedure had been properly reviewed and approved. The inspectors also witnessed the test and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors also verified that PMT activities were conducted in accordance with applicable work order (WO) instructions, or 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 CAP.

- Unit 2: PMT for B1 RHRSW Pump per 2-SI-4.5.C.I(3), RHRSW Pump and Header Operability and Flow Test
- Unit 2: PMT for B Emergency Diesel Generator Starting Air System per EPI-0-082-DGZ004, Diesel Generator B Redundant Start Test
- Unit 2: PMT for HPCI per 2-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test and Rated Reactor Pressure.
- Unit 2: PMT for RCIC per 2-SR-3.6.1.3.5, MOV Operability Test and 2-SR-3.5.3.3, RCIC System Rated Flow at Normal Operating Pressure.
- Unit 3: PMT for 3EA LPCI MG Set per WO 07-711926-001, step 1.6.
- Unit 2: PMT for RHR Div I per 2-SR-3.5.1.6, RHR Rated Flow Test
- Unit 2: PMT for Rx Zone Isolation Damper 2-FSV-064-0043 per 0-SR-3.6.4.2.1, Secondary Containment Isolation Valve Stroke Time

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 2 Scheduled Refueling Outage (U2C14)

a. Inspection Scope

During February 19 to March 31, 2007, 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 through the end of the reporting period. Activities occurring after March 31, 2007, will be documented in the next inspection report. Some of the more significant inspection activities conducted by the inspectors were as follows:

Outage Risk Assessment

Prior to the Unit 2 scheduled 42 day U2C14 refueling outage that began on February 20, the inspectors attended outage risk assessment team meetings and reviewed the Outage Risk Assessment Report to verify that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing an outage plan that assured defense-in-depth of safety functions were maintained. The inspectors also reviewed the daily U2C14 Refueling Outage Reports, including the Outage Risk Assessment Management (ORAM) Safety Function Status, and regularly attended the twice a day outage status meetings. 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 contingency response plan. Furthermore, the inspectors frequently discussed risk conditions and designated protected equipment with Operations and outage management personnel to assess licensee awareness of actual risk conditions and mitigation strategies.

Shutdown and Cooldown Process

The inspectors witnessed the shutdown and cooldown of Unit 2 in accordance with licensee procedures SPP-12.1, Conduct of Operations; 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations; and 2-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

Decay Heat Removal

The inspectors reviewed licensee procedures 2-OI-74, Residual Heat Removal System (RHR); 2-OI-78, Fuel Pool Cooling and Cleanup System; and Abnormal Operating Instruction 0-AOI-72-1, Alternate 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 Alternate Decay Heat Removal (ADHR) system. 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., tag order TO-2007-001, section 2-075-0003 for CS System I Inboard Injection Flow

Control Valve, 2-FCV-075-0025; and 2-TO-2007-0001, Section 2-064-0008, for work on Outboard Main Steam Isolation Valves (MSIV)); witnessed the tag removal for 2-TO-2007-0001, Sections 2-085-0025 and 2-085-0033 for the hydraulic control unit modules; and verified tags on amphenol disconnects revised to tagorder Section 2-85-0001B, tag nos. 4000 and 3997

- Verified Reactor Coolant System (RCS) inventory controls, especially during evolutions involving operations with the potential to drain the reactor vessel (OPDRV) controlled per 2-POI-200.5
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters (e.g., RCS pressure, level, flow, and temperature) and TS compliance during the various shutdown modes of operation, and mode transitions
- 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
- Routine tours of the control room, reactor building, refueling floor and drywell

Reactor Vessel Disassembly and Refueling Activities

The inspectors witnessed selected activities associated with reactor vessel disassembly, and reactor cavity flood-up and drain down in accordance with 2-GOI-100-3A, Refueling Operations (Reactor Vessel Disassembly and Floodup). Also, on numerous occasions, the inspectors witnessed fuel handling operations during the three Unit 2 reactor core fuel shuffles performed in accordance with TS and applicable operating procedures, such as GOI-100-3A, Refueling Operations (In Vessel), GOI-100-3B, Operations in the Spent Fuel Pool, and GOI-100-3C, Fuel Movement Operations During Refueling. The inspectors verified specific fuel movements as delineated by the Fuel Assembly Transfer Sheets (FATF). Furthermore, the inspectors also witnessed and examined the video verification of the final completed reactor core conducted per Attachment 6, of GOI-100-3C.

Torus Closeout

The inspectors performed a detailed closeout inspection of the Unit 2 suppression chamber on March 28.

Corrective Action Program

The inspectors reviewed PERs generated during U2C14 and attended management review committee (MRC) meetings to verify that initiation thresholds, priorities, mode holds, operability concerns and significance levels were adequately addressed. Resolution and implementation of corrective actions of several PERs were also reviewed for completeness.

b. Findings

Introduction: The inspectors identified a unresolved item associated with an inadequate procedure and failure to implement a procedure for installing a freeze seal.

Description: On February 23, 2007, the inspectors observed a licensee contractor performing freeze seal activities to support replacing 2-DRV-010-505, Unit 2 Reactor Vessel Bottom Drain to the Reactor Water Cleanup System. This was a normally open 2-inch manual valve which was non-insoluble from the Reactor Vessel. A single freeze seal was made on both the upstream and downstream side of the valve, in order to cutout and weld in a replacement valve. During the freeze seal evolution, the inspectors observed a number of problems associated with the freeze seal procedure itself and the actual freeze seal evolution. The specific MSI-0-000-PLG001 procedural compliance problems witnessed by the inspectors were as follows: failure to use specified Personal Protection Equipment (face shield, apron, gloves, etc.); failure to document freeze seal temperatures at 5 minute intervals; failure to ensure that all additional liquid nitrogen makeup bottles would fitup to the freeze seal system; and failure to maintain a continuous supply of liquid nitrogen to the freeze seal jacket. The specific MSI-0-000-PLG001 procedural deficiencies noted by the inspectors were as follows: lack of guidance for ensuring availability of backup bottles; and lack of contingency plans in case of a loss of the freeze plug. Plant conditions at the time of the freeze seal evolution were as follows: Reactor Vessel Head installed, but not fully tensioned (de-tensioning in progress), Division I of emergency cooling water systems out of service for outage activities, and high decay heat load.

In particular, while the inspectors were observing freeze seal operations, the contracted freeze seal operator noticed that the liquid nitrogen bottle connected to the upstream freeze seal (i.e., unisolable side connected to the reactor vessel bottom drain) had become exhausted. And although the operator did have a non-procedurally required backup bottle of nitrogen hooked up, this bottle was also determined to be empty. This condition resulted in a total loss of liquid nitrogen supply to the freeze seal jacket contrary to step 7.2[14] of MSI-0-000-PLG001. The operator promptly called for a replacement bottle, however contrary to step 3.0.L the fittings on this bottle were incompatible with the freeze seal equipment. A second replacement bottle was then requested which did fitup to the freeze seal apparatus, and the operator was able to restore liquid nitrogen flow. The loss of liquid nitrogen coolant flow to the upstream freeze seal only lasted about 5 -10 minutes. During this time the freeze seal jacket temperature increased to match the plug temperature, and frost band began to deteriorate. However, the freeze plug temperatures did not significantly change due to the short duration without a liquid nitrogen supply.

Furthermore, several days after 2-DRV-010-505 valve was replaced, and the freeze seal equipment was removed, the inspectors reviewed MSI-0-000-PLG001 and noticed that none of the procedure steps after 7.2[13] thru [16], and 7.3[1] thru [10.1] were signed off as complete.

Analysis: This finding was considered to be greater than minor because it is associated with the Barrier Integrity cornerstone attributes of Human Performance and Procedure Quality, and adversely affected the cornerstone objective to provide reasonable assurance that the Reactor Coolant System barrier provided protection to the public from radio nuclide releases caused by accidents or events. The inspectors evaluated the finding using IMC 0609, Appendix G, Shutdown Operations Significance Determination Process (SDP). According to Figure 1 and Checklist 6, of Appendix G, a

Phase 3 Analysis was required. The safety significance of the finding remains to be determined until completion of the Phase 3.
Enforcement: Criterion V, Instructions Procedures and Drawings, of 10 CFR 50, Appendix B, requires, in part, that, activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with the instructions. Contrary to this, quality procedure MSI-0-000PLG001, Installation of Freeze Seals, was inadequately and incompletely implemented, and lacked important guidance, which led to a degraded condition that challenged the integrity of an unisolable freeze seal on the Reactor Vessel bottom drain. The licensee entered this issue into their corrective action program as PERs 1120928 and 121179. Pending completion of the Significance Determination Process analysis, this item is considered as an unresolved item: URI 50-260/2007002-01, Failure to Follow the Freeze Seal Procedure and Procedural Inadequacy.

.2 Unit 2 Forced Outage Due To Automatic Scram

a. Inspection Scope

On January 11, 2007, Unit 2 entered an unplanned forced outage due to an automatic reactor scram. The inspectors examined the conduct of critical activities associated with the Unit 2 forced outage pursuant to TS, applicable procedures, and the licensee's outage risk assessment and management plans. The unit was restarted on January 12, and returned to full power operation on January 16. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of Hot Shutdown conditions, and critical plant parameters
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities
- Reactor Startup and Power Ascension activities

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 2 forced outage to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required. Certain aspects of the resolution and implementation of corrective actions of several PERs were also examined and/or verified.

b. Findings

No findings of significance were identified.

.3 Unit 3 Forced Outage Due To Automatic Scram

a. Inspection Scope

On February 9, 2007, the inspectors examined critical activities associated with the unplanned Unit 3 forced outage, caused by an automatic scram, to verify that they were conducted in accordance with TS, applicable procedures (e.g., 3-GOI-100-3A, Reactor

Startup), and the licensee's outage risk assessment and management plans. Unit 3 restarted on February 12 and returned to full power operation on February 14. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of Hot Shutdown conditions, and critical plant parameters
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities
- Drywell closeout inspection
- Reactor Startup and Power Ascension activities

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. Certain aspects of the resolution and implementation of corrective actions of several PERs were also examined and/or verified.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

- 2-SI-3.3.6, ASME Section XI System Pressure Test of the Core Spray System (ASME Section III Class 2)
- 2-SR-3.5.1.6(CSII), Core Spray Flow Rate Loop II**
- 2-SR-3.3.1.1.5, Source Range Monitor and Intermediate Range Monitor Overlap Verification
- 1/2-SI-4.9.A.1.d(B), Emergency Diesel Generator B Two Year Inspection
- 3-SR-3.8.1.6, Common Accident Signal Logic
- 2-SI-4.7.A.2.g-2/FHa, Unit 2 Drywell Head Local Leakrate Test (LLRT)*
- 2-SI-4.7.A.2.g-3/76f, Unit 2 Containment Inerting System: Penetration X-52C LLRT*

* Containment isolation valve
** IST

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 reviewed. 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 program activities; and assessed worker exposures to radiation and radioactive material.

Radiological postings and material labeling were directly observed during tours of the Unit 1, Unit 2, and Unit 3 turbine and reactor buildings; Unit 2 drywell; 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 radioactive material stored in the spent fuel pools. The inspectors also reviewed selected Radiological Control (Radcon) procedures and radiation work permits (RWPs), and discussed current access control program implementation with Radcon supervisors.

During the inspection, radiological controls for work activities in High Radiation Areas (HRA) were observed and discussed. The inspectors attended pre-job briefings for work performed on the 2C RHR Heat Exchanger floating head and directly observed the work activities involved. Inspectors also observed Unit 2 Control Rod Drive work, Unit 2 Reactor Water Clean-Up (RWCU) Heat Exchanger encapsulation tasks, and various ongoing maintenance activities within the Unit 2 drywell. The inspectors observed workers' adherence to RWP guidance and Health Physics Technician (HPT) proficiency in providing job coverage. 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 the drywell. 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.

Radcon program activities were evaluated against 10 CFR Part 20; Technical Specification (TS) Sections 5.4, Procedures, and 5.7, HRA; Regulatory Guide 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 the Attachment to this inspection report (IR).

Problem Identification and Resolution Multiple Problem Evaluation Reports (PERs) and two audits associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with Radcon supervisors. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with licensee procedure Standard Programs and Processes (SPP) 3.1, Corrective Action Program, Revision (Rev.) 11. Specific documents reviewed are listed in the Attachment to this IR.

The inspectors completed 21 of the specified line-item samples detailed in Inspection Procedure (IP) 71121.01.

b. Findings

Introduction. Two examples of a Green self-revealing NCV of 10 CFR 20.1501(a)(2)(i) were identified for failure to conduct surveys that were reasonable, under the circumstances, to evaluate the magnitude and extent of radiation levels in areas where work was performed.

Description. On February 9, 2007, a Unit 1 maintenance electrician received an electronic dosimeter (ED) dose rate alarm while working on the Unit 1 Fuel Pool Cooling Heat Exchanger. The electrician was performing work in an area that was not posted as an HRA, and the worker had not received the Health Physics (HP) briefing required to enter an HRA. Upon investigation, HP staff identified dose rates in the work area of approximately 200 millirem/hour (mrem/hr) that had not been identified during a previous survey. HP staff immediately posted and controlled the work area as an HRA. The issue was entered into the corrective action program as PER 119482.

On February 16, 2007, a carpenter received an ED dose rate alarm of 253 mrem/hr while building scaffold around Residual Heat Removal (RHR) piping on top of the Unit 2 torus. Pre-job surveys showed whole-body dose rates in the area to be only 25 mrem/hr. Upon investigation, HP staff identified general area dose rates of approximately 250 mrem/hr in the work area, due to hot spots on the RHR piping. HP immediately posted and controlled the area as an HRA. The issue was documented in the licensee's corrective action program as PER 119829.

Analysis. The failure to conduct surveys that were reasonable under the circumstances to evaluate the magnitude and extent of radiation levels in work areas is a performance deficiency. The finding is more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process, and it adversely affected the cornerstone objective because the failure to conduct adequate radiation surveys did not ensure adequate protection of worker health and safety from exposure to radiation. The finding was evaluated using the Occupational Radiation Safety Significance Determination Process (SDP), and was determined to be of very low safety significance. The failures to survey did not pose a substantial potential for overexposure and did not affect the ability to assess doses. The involved individuals were monitored for exposures from external radiation fields using alarming electronic dosimeters. When the dose rate alarms actuated, the workers promptly left the area before accruing any significant exposure.

The cause of the finding was directly related to the work activity coordination aspect of the human performance cross-cutting area because pre-job radiological surveys were inadequate to apprise personnel of plant conditions that affected work activities.

Enforcement. 10 CFR 20.1501(a)(2)(i) requires, in part, that each licensee shall make surveys that are reasonable, under the circumstances, to evaluate the magnitude and extent of radiation levels. Contrary to this, the licensee failed to identify HRAs in work locations before sending employees to perform work in those areas. Because this violation was of very low safety significance and was entered into the licensee's corrective action program, it is being treated as NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-259, 260, 296/2007002-02, Two Examples of Failure to Perform Adequate Surveys.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) The inspectors evaluated ALARA program guidance and implementation for ongoing tasks associated with the U2C14 refueling outage. In addition, post-outage ALARA activities associated with the Unit 3 Cycle 12 (U3C12) refueling outage and ALARA planning and performance for recovery efforts on Unit 1 were evaluated. The inspectors reviewed and discussed with licensee staff ALARA work plan documents, including dose estimates and prescribed ALARA controls for selected outage work activities expected to incur significant collective doses. The inspectors reviewed the implementation of dose-reduction initiatives for high person-rem expenditure tasks. These elements of the ALARA program were evaluated for consistency with the methods and practices delineated in applicable licensee procedures.

The implementation and effectiveness of ALARA planning and program initiatives during work in progress were evaluated. The inspectors made direct field observations of Unit 2 work activities involving electrical disconnection of recirculation pump motors, chemical decontamination preparation, replacement of a control rod drive (CRD) accumulator discharge valve and work supporting RHR heat exchanger repairs. Observations of various work in the turbine building in support of the extended power uprate project and a small amount of work being performed to support the unit 1 recovery were also made. The inspectors interviewed radiation workers and HP staff to assess their understanding of dose reduction initiatives and their current and expected final accumulated occupational doses at completion of the task.

Projected RWP dose expenditure estimates from U2C14 and Unit 1 recovery efforts were compared to actual dose expenditures, and noted differences were discussed with cognizant ALARA staff. Changes to dose budgets relative to changes in job scope were identified and discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. The inspectors reviewed the plant's source-term reduction initiatives that had been applied to Unit 1 as well as the initiatives proposed to be instituted in Units 2 and 3 over the next five years. The potential causes of the elevated source terms experienced over the last cycle were discussed with both ALARA and Chemistry department personnel. The adverse impact of the reduction of zinc addition to increase fuel integrity on drywell dose rates was discussed with Chemistry department personnel. The licencee's performance of a chemical decontamination to reduce drywell dose rates was discussed with ALARA personnel. The effectiveness of selected shielding packages installed for the current outage was assessed through completion of independent radiation surveys and comparison to applicable licensee survey records and expected planning data. The licensee's use of X-ray fluorescence to measure residual elemental cobalt contamination following maintenance was discussed with Radiation Protection technicians.

The plant collective exposure histories for calendar years (CY) 2003, 2004 and 2005, taken from data reported to the NRC pursuant to 10 CFR 20.2206, were reviewed and discussed with licensee staff, as were established goals for reducing collective exposure. The inspectors reviewed the applicable guidance and examined dose records of declared pregnant workers during CY 2005 and 2006 to evaluate current gestation doses for declared pregnant workers.

ALARA activities were evaluated against the requirements specified in 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. In addition, licensee performance was evaluated against Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable, and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Sections 2OS1 and 2OS2 of the Attachment to this IR.

<u>Problem Identification and Resolution</u> Licensee corrective action documents associated with ALARA activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with the corrective action program. Specific self-assessments, audits, and PERs reviewed and evaluated in detail for this inspection area are identified in Section 2OS2 of the Attachment to this IR.

The inspectors completed 29 of the specified line-item samples detailed in IP 71121.02.

b. <u>Findings</u>

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Current licensee programs for monitoring, tracking, and documenting the results of both routine and abnormal liquid releases to onsite and offsite surface and ground water environs were reviewed and discussed in detail. The inspectors reviewed and discussed the effect of routine effluent liquid releases made in accordance with the Offsite Dose Calculation Manual (ODCM) requirements on surface water concentrations and their potential impact on tritium recently identified in samples collected from onsite groundwater monitoring wells. The inspectors toured established well locations, and discussed current monitoring activities and analysis results. Reports associated with abnormal liquid releases and corrective actions were reviewed and discussed with responsible licensee representatives to evaluate the potential onsite/offsite environmental impact of significant leakage/spills from onsite systems, structures, and components. Licensee routine surveillances to rapidly identify and mitigate any abnormal leaks from liquid radioactive waste tanks, processing lines, and spent fuel pools were reviewed and discussed in detail.

The inspectors completed two of the specified radiation protection line-item samples detailed in IP 71122.01.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization The inspectors reviewed the plant's solid radioactive waste (radwaste) system description in the Browns Ferry Nuclear Plant FSAR and process control program (PCP). The most recent radiological effluent release report was reviewed for information on the types and amounts of waste disposed. The scope of the licensee's audit program was reviewed to verify that it met the requirements of 10 CFR 20.1101. The inspectors walked down the accessible portions of the liquid and solid radioactive waste processing systems to verify and assess that the current system configuration and operation agreed with the FSAR and PCP. The use of video cameras to monitor batch tank levels was also discussed with the radwaste and operations personnel.

The inspectors reviewed the radiological operating report for any documented changes to the radwaste processing systems and discussed the observations with radwaste and operations personnel. The inspectors reviewed the plant's process for transferring radioactive resin and sludge discharges into shipping/disposal containers to determine if appropriate waste stream mixing and/or sampling procedures and methodology for waste concentration averaging provided representative samples of the waste product for waste classification purposes. The inspectors reviewed current 10 CFR 61 analysis results and the procedures for obtaining the samples to support the analysis. The scaling factors used for radioactive waste streams and calculations used for determining the amount of hard to detect nuclides were reviewed. The program was reviewed to verify compliance with 10 CFR 61.55-56 and Appendix G of 10 CFR 20.

The inspector reviewed the program for provisions that would ensure that the waste stream composition accounted for changes in operational parameters and would remain valid between required periodic updates.

Transportation The inspectors observed the preparation and shipment of resin to a vendor facility. The observations included packaging, surveying, labeling, placarding, vehicle checks, driver's briefing and emergency instructions, a review of shipping papers provided to the driver, and licensee final verification of shipment readiness. The inspectors were unable to observe a Type B shipment as none were scheduled during the inspection period. The inspectors reviewed shipping documentation for several shipments that had occurred in the previous year. The inspectors reviewed the Quality Assurance (QA) surveillance documentation verifying compliance with the Certificate of Compliance for the Type B packages that included irradiated control rod blades and drives. The inspectors observed, interviewed, and reviewed the training records of the radwaste workers who were involved in the shipments.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed are listed in the Attachment to this IR.

Problem Identification and Resolution Select PERs and self assessments were reviewed in detail and discussed with cognizant licensee personnel. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1, Corrective Action Program, Rev. 11. Documents reviewed are listed in the Attachment to this report.

The inspectors completed six of the specified line-item samples detailed in IP 71122.02.

b. Findings

Introduction. A Green self-revealing NCV of 10 CFR 71.5 was identified involving two shipments of radioactive material that exceeded the regulatory dose limits upon arrival at a vendor facility. The violation involved a failure to properly package material such that, under conditions normally incident to transportation, the radiation levels at the external surface of the package would not exceed applicable Department of Transportation (DOT) limits.

Description. On April 20, 2005, the licensee made two shipments, one containing three boxes (packages) and another containing four boxes of contaminated equipment from its Browns Ferry facility to a vendor's facility located in Lynchburg, Virginia. The shipments (numbers (Nos.) 050423 and 050424) consisted of contaminated equipment and were shipped as radioactive material, surface contaminated object (SCO-II). The total activity of both shipments was 209 millicuries of solid/metal oxides. The licensee's radiation survey performed prior to shipping indicated that the maximum radiation level on any external surface of the packages was 190 millirem per hour (mrem/hr). When the shipments arrived at the vendor's facility on April 21, 2005, a receipt radiation survey performed by the vendor indicated that maximum contact radiation dose rates on the external surface of two packages exceeded 200 mrem/hr using a Geiger-Mueller (GM) radiation survey instrument. In each case, the vendor identified a small area on the bottom of the package where the dose rates exceeded the limit. The vendor identified one box in Shipment No. 050423 with a contact dose rate of 280 mrem/hr and one box in Shipment No. 050424 with a contact dose rate of 300 mrem/hr. The instrument used by the vendor was within its calibration due date. The vendor informed the licensee of this condition on April 21, 2005.

On April 29, 2005, representatives from Browns Ferry performed confirmatory radiation surveys on the shipments using GM survey instruments and obtained contact dose rates of 220 and 250 mrem/hr for the same boxes as identified by the vendor to have exceeded the 200 mrem/hr dose rate limit. The source of the elevated dose rates was determined to be hot particles. While the cause of the difference between the vendor's measurements (280 and 300 mrem/hr) and the licensee's measurements (220 and 250 mrem/hr) could not be explained with certainty, the inspectors noted possible movement of the hot particles from April 20 to April 29, 2005, could have affected the dose rate measurements. Based on the small size and high specific activity, a small change in the particle's distance to the instrument could have a large impact on the dose rate measured. Additionally, there is an allowed tolerance in the calibration of the instruments. The licensee established additional supervisory review and approval prior to shipping packages approaching DOT limits and entered the issue into their corrective action program as PER 81364.

Analysis. The licensee's failure to ensure radiation levels did not exceed applicable DOT dose rate limits under conditions normally incident to transportation is a performance deficiency because compliance with the requirement was reasonable and within the licensee's ability to achieve. This finding is more than minor because it is associated with the Plant Facilities/Equipment and Instrument attribute of the Public Radiation Safety cornerstone and it adversely affected the cornerstone objective in that the improper transportation packaging resulted in a shipping container with external dose levels exceeding regulatory requirements. The significance of this finding was evaluated using the Public Radiation Safety Significance Determination Process, and was determined to be of very low safety significance (Green) because the area on the packages with elevated radiation levels were inaccessible to the public and the radiation levels were less than two times the DOT limit.

Enforcement. 10 CFR 71.5 requires each licensee who transports licensed materials on public highways to comply with the requirements of the DOT regulations in 49 CFR Parts 170 through 189. The 49 CFR 173.441(a), "Radiation Level Limitations," requires that each package of radioactive material offered for transportation be designed and prepared for shipment so that, under conditions normally incident to transportation, the radiation level does not exceed 200 mrem/hr at any point on the external surface of the package.

Contrary to the above, on April 20, 2005, Browns Ferry Nuclear Station shipped radioactive materials to a vendor's facility located in Lynchburg, Virginia, but failed to properly prepare the shipment for transport in that on April 21, 2005, when the shipment arrived at the vendor facility, the vendor measured radiation levels of 280 and 300 mrem/hr on portions of the external surface of two packages, which exceeded the DOT regulatory limit. However, because this finding is of very low safety significance, and has been entered into the licensee's corrective action program as PER 81364, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 050-259, 260, 296/2007002-03, Failure to Properly Prepare a Radioactive Material Package for Shipment.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Barrier Integrity Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the Performance Indicators (PI) listed below, including procedure SPP-3.4, Performance Indicator for NRC Reactor Oversight Process for Compiling and Reporting PI's to the NRC. The inspectors reviewed the raw data for the PI's listed below for the first through fourth quarter of 2006. The inspectors compared the licensee's raw data against graphical representations and specific values reported to the NRC in the fourth quarter 2006 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 chemistry and engineering personnel to discuss and go over licensee records to verify that the PI data was appropriately captured, calculated correctly, and discrepancies resolved. The inspectors reviewed Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, to verify that industry reporting guidelines were applied.

- RCS Activity for Units 2 and 3
- RCS Leakage for Units 2 and 3

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety and Public Radiation Safety Cornerstones

a. Inspection Scope

The inspectors sampled licensee records to verify the accuracy of reported data for the PI's listed below. To verify the accuracy of the reported PI elements, the reviewed data were assessed against guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 4, and the PI Frequently Asked Questions list.

Occupational Exposure Control Effectiveness - The inspectors reviewed the PI results for the period of January 2006 through February 2007. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and licensee procedural guidance for collecting and documenting PI data. Report section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas and the review of related PERs. Documents reviewed are listed in the Attachment to this report.

RETS/ODCM Radiological Effluent - The inspectors reviewed the PI results for the period of January 2005 through February 2007. For the assessment period, the inspectors reviewed cumulative and projected doses to the public. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4OA2 Identification and 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 and regularly attended daily management review committee meetings. 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

No findings of significance were identified.

.2 Focused Annual Sample Review

a. Inspection Scope

The inspectors reviewed the corrective actions associated with PERs 114913 and 114967. These PERs were initiated to address inspector identified issues related to Station Blackout (SBO) mitigation documented in inspection report (IR) 50-260, 296/2006005. As part of this focused inspection, the inspectors reviewed Revisions 65 through 69 of 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout, and examined licensed operator training on the new SBO mitigation strategies. The inspectors also witnessed two separate dual-unit simulator demonstrations of the revised AOI-57-1A mitigation strategies with actual operating crews on January 23 and February 14, 2007. Furthermore, the inspectors met with the Operations Superintendent and Training supervision to discuss and critique the results of these unsuccessful simulator demonstrations.

Following these unsuccessful simulator demonstrations of the worst case Unit 2 SBO event, which causes a loss of all EECW, the licensee initiated another PER 119778 and developed a new more simplified mitigation strategy based on cross-tieing Unit 1 and 2 4KV shutdown boards. The inspectors reviewed the new mitigation strategy as incorporated into AOI-57-1A, Revison 70, and also reviewed the associated training notice issued on March 12, 2007. Lastly, the inspectors witnessed a third and successful simulator demonstration for the worst case Unit 2 scenario on April 9.

b. Findings and Observations

After it became apparent that the initial changes to AOI-57-1A were only partially successful, the licensee took additional corrective actions to develop a new simplified mitigating strategy that would ensure an adequate cooling water supply to the running EDGs could be maintained or restored in a timely manner under all SBO conditions.

Introduction: The inspectors identified a Green NCV of 10CFR50, Appendix B, Criterion XVI, for ineffective corrective actions to ensure that the operating EDGs during an Unit 2 SBO event would have sufficient cooling water under worst case licensing-basis conditions. The licensee was successful in demonstrating their ability to restore adequate cooling water to the operating EDGs during a worst case Unit 1 or 3 SBO event.

Description: After witnessing two unsuccessful simulator demonstrations of the operators' ability to execute the licensee's revised SBO mitigation strategy, the inspectors concluded that the licensee's corrective actions to address NCV 5000260, 296/2006005-01, Lack of Assured Cooling Water for Emergency Diesel Generators During SBO Conditions, were ineffective for Unit 2. Even with significant changes to AOI-57-1A, LOOP/SBO, and additional operator training, the operators were unable to demonstrate their capability to mitigate the worst case SBO event in accordance with the licensing basis for an Unit 2 SBO. The operators were unable to execute the revised mitigating strategies of AOI-57-1A for restoring cooling water to the operating EDGs within the required seven minute timeline before the operating EDGs would overheat.

Analysis: This finding was considered to be greater than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences. The SDP Phase 1 analysis did not screen to Green, consequently an SDP Phase 2 analysis was required. With no credit for operators being able to crosstie 4KV shutdown boards to power up an EECW pump or align a swing RHRSW/EECW pump, within the prescribed AOI-57-1a timeline, the Phase 2 analysis for a LOOP event did not screen to Green. A Phase 3 risk analysis was performed by a regional Senior Reactor Analyst.

Input from the NRC's plant-specific risk model and manual calculations to represent the likelihood of the specific combinations of EDGs that represent the finding were used in a manual risk calculation. Because the sequences involved SBO scenarios, the Large Early Release Frequency (LERF) metric was used. The developed sequences involved a loss of offsite power, followed by a common cause failure of either 3 or 4 EDGs. Only very specific combinations of operating EDGs would result in the loss of cooling water to all EDGs. The common cause failure rates were reduced by the ratio of those specific combinations to the total combinations for each failure of interest. The analysis assumed that one running EECW pump would provide sufficient EDG cooling for a long enough period that some operator recovery credit was possible. However, conservatively no operator recovery credit was allowed for the condition where no cooling water pumps were aligned to the EDGs. The conditional core damage probability (CCDP) for SBO was calculated using the NRC's Standardized Plant Analysis Risk (SPAR) model for Browns Ferry 2. This CCDP was multiplied by the other factors to obtain the change in risk from the finding. The analysis resulted in a large early release frequency of less than 1E-7 per year, which is of very low safety significance. The finding was determined to be of very low safety significance because of the low frequency of occurrence of the specific combination of multiple EDG failures that could lead to a loss of cooling water flow to all of the running EDGs.

The cause of finding was directly related to the appropriate and timely corrective action aspect of the Problem Identification and Resolution cross-cutting area because corrective actions developed for Unit 2 SBO mitigation strategy deficiencies were not effective in ensuring timely restoration of cooling water to the EDGs

Enforcement: Criterion XVI, Corrective Action, of 10 CFR 50, Appendix B, requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to Criterion XVI, the licensee failed to implement effective corrective actions for Unit 2 to resolve the condition adverse to quality identified by NCV 5000260, 296/2006005-01. However, because this failure to implement effective corrective actions is considered to be of very low safety significance, and has been entered into the licensee's corrective action program as PER 119778, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000260/2007002-04, Inadequate Corrective Actions to Resolve Deficiencies in SBO Mitigation Capabilities.

4OA3 Event Follow-up

.1 Unit 2 Reactor Scram

a. Inspection Scope

On January 11, 2007, the Unit 2 reactor automatically scrammed from 100% power due to a main turbine generator (MTG) load reject signal that occurred during the performance of Operating Instruction 2-OI-47, Main Generator [EL] Voltage Control Rheostat Test. The load reject signal was caused by a relay failure in the MTG voltage regulator during testing as the regulator was taken from the automatic to manual mode. The inspectors promptly responded to the control room and verified that the unit was in a stable Mode 3 (Hot Shutdown) condition, and confirmed that all safety-related mitigating systems and automatic functions operated as designed. The inspectors evaluated safety equipment and operator performance before and after the event by examining existing plant parameters, strip charts, plant computer historical data displays, operator logs, the alarm typewriter Sequence of Events printout, and the critical parameter trend charts in the post-trip report. The inspectors also interviewed responsible onshift Operations personnel, examined the implementation of applicable annunciator response procedures (ARP), AOIs, and EOIs, including 2-AOI-100-1, Reactor Scram. Furthermore, the inspectors reviewed and verified that written notification made in accordance with 10 CFR 50.72. The inspectors subsequently discussed the preliminary cause of the apparent relay failure with responsible Maintenance and Engineering personnel.

b. Findings

No significant findings were identified.

.2 Unit 3 Reactor Scram

a. Inspection Scope

On February 9, 2007, Unit 3 experienced an automatic reactor scram from 100 percent power due to low-low reactor water level from a loss of condensate/feedwater flow. The loss of condensate flow was caused by an inadvertent isolation of the condensate full flow demineralizers during ongoing online software modifications of the condensate and demineralizer water system control logic. The inspectors promptly responded to the control room and verified that the unit was in a stable Mode 3 (Hot Shutdown) condition. The inspectors also confirmed that all safety-related mitigating systems and automatic functions operated properly. The inspectors evaluated safety equipment and operator performance before and after the event by examining existing plant parameters, strip charts, plant computer historical data displays, operator logs, the alarm typewriter Sequence of Events printout, and the critical parameter trend charts in the post-trip report. The inspectors also interviewed responsible onshift Operations personnel, examined the implementation of applicable ARPs, AOIs, and EOIs, particularly 3-AOI-100-1, Reactor Scram. Furthermore, the inspectors reviewed and verified that written notification was made in accordance with 10 CFR 50.72. The inspectors subsequently discussed the preliminary cause of the loss of condensate flow with responsible Operations and Engineering personnel.

b. Findings

No significant findings were identified.

4OA5 Other

.1 Review of Institute of Nuclear Power Operations (INPO) and World Association of Nuclear Operators (WANO) Reports

The inspectors reviewed the following INPO and WANO reports:

- Browns Ferry Unit 1 Startup Readiness Review (Final Report - September 2006)
- Browns Ferry Nuclear Plant WANO Peer Review (Interim report - December 1, 2006)

These reports did not identify any safety or risk significant issues that had not been previously recognized and/or examined by the NRC.

.2 Instrumentation and Control Mechanics Work Hours In Excess of Overtime Limits

Introduction: A Green NCV was identified by the inspectors for multiple instances of key maintenance plant personnel exceeding the overtime limits of TS 5.2.2.d without prior authorization by the Plant Manager.

Description: In January 2007, during routine plant status review, the inspectors became aware that the Instrumentation and Control mechanics (ICMs) were working a routine schedule in excess of 72 hours in a seven day period, without a pre-approved deviation from the Plant Manager. Section 3.B.2 of the SPP procedure 1.5, Overtime Restrictions (Regulatory), restricted the work hours of key maintenance personnel to 72 hours in a seven day period. However, since October 2006, ICMs on dayshift had been regularly working a two week schedule of seven 12.5 hour days, followed by five eight hour days, with two days off. Even after discounting turnover time, the ICM working hours were significantly greater than the allowed 72 hours in a seven day period. Once this issue was presented to maintenance management, it became apparent to the inspectors that there were certain fundamental misunderstandings by management regarding the exclusion of turnover time and lunch time from the overtime limits. In the first case, maintenance management was discounting two hours a day for turnover time even though less than an hour of this time actually involved what would be considered turnover-related activities. Secondly, management was discounting a half hour a day for lunch, even though this was expressly disallowed by SPP-1.5. After being made aware of these discrepancies, Maintenance management promptly rescheduled the ICM work hours to comply with the SPP-1.5 overtime limitations. This finding was caused by inadequate management oversight and awareness of the administrative requirements of SPP-1.5.

Analysis: Failure to authorize overtime in accordance with TSs was a performance deficiency and a finding. The finding was considered to be greater than minor because if left uncorrected it could become a more significant safety concern due to excessive fatigue by personnel performing safety-related activities. However, this finding was determined to a be a non-SDP miscellaneous finding and considered to be of very low safety significance (Green) because no specific performance deficiencies were identified for the individuals during the time they exceeded the established overtime limits. This non-SDP finding was reviewed by NRC management.

Enforcement: Technical Specification 5.2.2.d requires administrative procedures be developed and implemented to limit the working hours of personnel who perform safety-related functions. The TS also require that any deviations from the established overtime limitations shall be authorized in advance by the Plant Manager in accordance with approved administrative procedures, with documentation of the basis for granting the deviation. The licensee’s program for controlling overtime to meet TS 5.2.2.d was established by SPP-1.5, Overtime Restrictions (Regulatory). Contrary to the TS 5.2.2.d and SPP-1.5, multiple instances were identified where I&C maintenance personnel performing safety-related work were working in excess of the overtime limits established by SPP-1.5 without obtaining prior authorization. However, because this failure to adequately control the overtime of key maintenance personnel is considered to be of very low safety significance and has been entered into the licensee's corrective action program as PER 119016, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000260, 296/2007002-05, Work Hours for I&C Mechanics Exceeded Overtime Limits Without Prior Authorization.

.3 Operation of Unit 2 Outside the Limits Allowed by the Power-Flow Map

Introduction: The inspectors identified a Green NCV of TS 5.4.1.a for failing to maintain Unit 2 core flow within the bounds of the Core Power/Flow Map established by 2-OI-68, Reactor Recirculation System.

Description: On February 5, 2007, during a routine tour of the Unit 2 control room, the inspectors observed that Unit 2 reactor recirculation core flow was greater than the maximum flow allowed by the Core Power/Flow Map. At that time, Unit 2 was in an end-of-cycle (EOC) coast down condition at 91% power while operating in the increased core flow (ICF) region of the Power/Flow Map. The maximum allowed core flow in the ICF region for 91% power was 106.1% core flow. Operators were controlling core flow at the extreme limit of the ICF region in order to maximize reactor power. However, actual core flow as observed by the inspectors was oscillating between 105.7% to 106.8%. Operators had inadvertently allowing core flow to increase slightly beyond the ICF limit for the existing power level. Upon notification, that Unit 2 core flow appeared to be outside the allowed limits of the Power/Flow Map, the US conferred with the on-call reactor engineer and promptly directed the operators to reduce reactor recirculation pump flow. The inspectors subsequently met with Operations management to discuss the lack of attention and familiarity of the Unit 2 operators in controlling and maintaining core flow within the prescribed limits of the Power/Flow Map while operating in the ICF region during EOC conditions. Operations management immediately initiated a PER, and issued an Operations Daily Instruction (ODI) to the shift crews. This ODI reinforced management expectations regarding the critical nature of monitoring and maintaining core flow within the ICF region.

Analysis: This finding was considered to be more than minor because if left uncorrected, operators could have unknowingly allowed core flow to exceed the analytical envelope of the fuel vendor's reload report transient analysis which would have been a more significant safety concern. The finding was associated with the Barrier Integrity cornerstone. In accordance with Phase 1 of the SDP, the finding was determined to be of very low safety significance because it is associated with fuel barrier integrity. Furthermore, core flow was still within the envelope of the fuel vendor's analytical limits and none of the reactor fuel thermal limits were exceeded.

The cause of the finding was directly related to the procedure compliance aspects of the Human Performance cross-cutting area because of inadequate communication of management and supervisory expectations for unit operations in the increased core flow region and lack of operator attention to the proceduralized power/flow map limits.

Enforcement: Technical Specifications 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A. Items 2g and 4a, of Appendix A, require procedures be established and implemented for general power operation, and specifically for the recirculating system. Contrary to the above, operators inadvertently allowed core flow to exceed the prescribed limits of the OI-68, Illustration 1, Unit 2 Power/Flow Map. However, because this finding is of very low safety significance and has been entered into the licensee's corrective action program as PER 119305, this violation is being treated as an NCV in accordance with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000260/2007002-06, Operation of Unit 2 Outside the Limits Allowed by the Power-Flow Map.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 5, 2007, the resident inspectors presented the integrated inspection results to Mr. Brian O'Grady, 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.

.2 Annual Assessment Meeting Summary

On April 2, 2007, the NRC's Senior Resident Inspectors and the Resident Inspectors assigned to the Browns Ferry Nuclear Plant, as well as the Region II Public Affairs Officer, met with the Tennessee Valley Authority (TVA) to discuss the NRC's Reactor Oversight Process (ROP) and the NRC's annual assessment of Browns Ferry's safety performance for the period of January 1, 2006, through December 31, 2006. The major topics addressed were: the NRC's assessment program, the results of Browns Ferry Units 1, 2 and 3 assessments, and future NRC inspection activities. Attendees included TVA management, site staff and members from the media and public.

This meeting was open to the public. The NRC's presentation material used for the discussion and list of attendees are available from the NRC's document system (ADAMS) as accession number ML070960080. ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations

The following findings of very low safety significance (Green) was identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Technical Specifications (TS) 5.7 requires that each high radiation area shall be barricaded and conspicuously posted as a high radiation area. On February 3, 2007, operations staff supplied extraction steam to the 2C2 high pressure feed water heater prior to HP staff reposting the room as a Locked High Radiation Area (LHRA). The door to the heater room was locked and secured, but the LHRA posting had been turned around so that the room was not posted in accordance with TS. The finding was of very low safety significance because it did not involve (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. The licensee entered this finding into its corrective action program as PER 96570.

- On February 9, 2007, shortly after the automatic scram of Unit 3, an operator restarted the 3B Recirculation Pump per 3-SR-3.9.3&4, Reactor Recirculation Pump Start Limitations. Subsequent review of the completed 3-SR-3.9.3&4 by the US several hours later determined that the operator had exceeded the allowed temperature limits of Technical Specifications Surveillance Requirement (TSSR) 3.4.9.4 which required recirculation loop temperature to be within 50°F of reactor pressure vessel coolant temperature. Contrary to TSSR 3.4.9.4, the temperature difference was 72°F. This finding is of very low safety significance because the resultant thermal stresses were not significant enough to adversely affect the reactor coolant pressure boundary. The licensee entered this finding into its corrective action program as PER 119489.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Berry, Systems Engineering Manager
T. Brumfield, Site Nuclear Assurance Manager
J. Burton, Design Engineering Manager
P. Chadwell, Operations Superintendent
J. Corey, Radiation Protection Manager
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
R. Davenport, Work Control and Planning Manager
J. DeDimenico, Asst. Nuclear Plant Manager
R. DeLong, Site Engineering Manager
A. Elms, Operations Manager
A. Feltman, Emergency Preparedness Supervisor
A. Fletcher, Field Maintenance Superintendent
J. Hopkins, Outage Scheduling Manager
R. Jones, General Manager of Site Operations
D. Langley, Site Licensing Supervisor
D. Matherly, Human Performance Manager
J. Mitchell, Site Security Manager
R. Rogers, Maintenance & Modifications Manager
B. O'Grady, Site Vice President
W. Pierce, Radioactive Waste Manager
D. Sanchez, Training Manager
E. Scillian, Operations Training Manager
C. Sherman, Radiation Protection Support manager
J. Sparks, Outage Manager
J. Steele, Outage Manager
J. Underwood, Acting Chemistry Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open

05000260/2007002-01	URI	Failure to Follow the Freeze Seal Procedure and Procedural Inadequacy (Section 1R20)

Opened and Closed

05000259, 260, 296/2007002-02	NCV	Two Examples of Failure to Perform Adequate Surveys (Section 2OS1)
05000259, 260, 296/2007002-03	NCV	Failure to Properly Prepare a Radioactive Materials Package for Shipment (Section 2PS2)

05000260/2007002-04	NCV	Inadequate Corrective Actions to Resolve Deficiencies in SBO Mitigation Capabilities (Section 4OA2.2)
05000260, 296/2007002-05	NCV	Work Hours for I&C Mechanics Exceeded Overtime Limits Without Prior Authorization (Section 4OA5.2)
05000260/2007002-06	NCV	Operation of Unit 2 Outside the Limits Allowed by the Power-Flow Map (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1R04; Equipment Alignment

Basis Documents

Technical Specifications, 3.8.1, AC Sources - Operating
Technical Specifications, 3.8.3, Diesel Fuel Oil, Lube Oil, Starting Air
Technical Specifications, 3.8.4, DC Sources - Operating
Final Safety Analysis Report, 8.5, Standby AC Power Supply and Distribution

Procedures

0-OI-82, Rev 91, Standby Diesel Generator System Operating Instruction
0-OI-18, Rev 49, Fuel Oil System Operating Instruction

Work Orders

05-725430-000, Air Start System Right Bank Clean and Inspect PM of Moisture Traps, Check Valves, Pilot Valves, and Selected Solenoid valves
06-711796-000, B Diesel Air Start System Routine PM
06-722242-000, Diesel Generator (DG) B Left Bank Air Compressor Worn Sheave Needs Replacement
06-723377-000, DG B Right Bank Starting Air System Air Dryer Does Not Run When Compressor is in Service
06-725107-000, DG B Left Bank Pressure Switch Alarm Sealed In With Pressure In Acceptable Range

Drawings

0-47E861-5, Rev 11, Flow Diagram, Cooling System and Lubricating Oil System, Standby Diesel Generator A
0-47E861-2A, Rev 6, Flow Diagram, Diesel Starting Air System, Diesel Generator B
0-47E840-3, Rev 02D, Flow Diagram, Fuel Oil System

Corrective Action Documents

105549, Diesel Center Information Station Fan 0-FAN-30-0074 Failed to Achieve Design Flowrate

Other Documents

OPL171.038, Rev 16, Diesel Generators and Standby Auxiliary Power System Training Lesson Plan
System Health Report, Systems 018/082/086, Diesel Generator, Diesel Air Start, and Fuel Oil, FY06P2

Section 1R08; Inservice Inspection Activities

Procedures

N-GP-18, Ultrasonic Testing Supplements, Rev. 14
N-UT-18, Manual Ultrasonic Examination Piping Welds and Vessels with Wall Thicknesses 2 Inches and Less that Are Not Required to be Examined in Accordance with Appendix VII, Rev. 27
N-UT-76, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, Rev. 6
N-MT-6, Magnetic Particle Examination for ASME and ANSI Code Components and Welds, Rev. 28
N-RT-1, Radiographic Examination of Nuclear Power Plant Components, Rev. 26

Corrective Action Documents (Condition Reports (CR)) and Action Items (AI)

119093, Piping Defect, 2/3/2007
103597, Crack in Weld on RWCU Decon Tap, 5/20/2006
120509, Unit 2 Steam Drier Tie Bar and Divider Plate Damage, 2/27/2007
80950, N-UT-66 Rev. Error, 4/19/2005
121224, UT Couplant, 3/9/2007 *

*CR created as a result of this NRC inspection

Section 1R12; Maintenance Effectiveness

Basis Documents

Technical Specifications, 3.6.3.2, Primary Containment Oxygen Concentration
Technical Requirements, 3.1.11, Hydrogen Monitoring Instrumentation
Technical Requirements, 3.6.2, Oxygen Concentration Monitors
Final Safety Analysis Report, 5.2.3.8, Containment Inerting System

Procedures

2/3-OI-76, Rev 61, Containment Inerting System

2/3-EOI-2, Rev 9, Primary Containment Control

Drawings

2/3-47E610-76-4, Rev 67, H2/O2 Analyzer Cabinet

Other Documents

System Health Report, System 076, Containment Inerting System, FY06P2
BFN-VTM-H124-0010, Rev 11, Hays-Republic / Div of Milton Roy Company, Oxygen and Hydrogen Analyzer Panels of the Containment Atmosphere Monitor System
BFN-VTD-H124-0020, Rev 2, Hays-Republic / Div of Milton Roy Company, Oxygen and Hydrogen Analyzers for Containment Atmosphere Monitor System
OPL 171.016, Rev 14, Primary and Secondary Containment Systems
MR Expert Panel Meeting Minutes, dated August 8, 2006
MR Expert Panel Meeting Minutes, dated September 21, 2006
SPP-6.6, Rev 8, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting
SPP-3.1, Rev 11, Corrective Action Program
0-TI-346, Rev 28, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting
Alarm Disablement Tech Evaluation and Screening, BFN Unit 2 Panel 2-9-7, 2-XA-55-7C, Window 22, Node 74-76-76
Alarm Disablement Tech Evaluation and Screening, BFN Unit 3 Panel 3-9-7, 3-XA-55-7C, Window 22, Node 74-76-86

2OS1: Access Controls to Radiologically Significant Areas

Problem Evaluation Reports (PERs)
PER 119482, Valid Dose Rate Alarm on RWP 07-110004, 2/09/07
PER 119829, Unanticipated dose rate alarm, 2/16/07
PER 96570, Miscommunication on locking of 2C2 heater room, 2/03/07
PER 121979, A submerged Tri-Nuc filter basket containing two used filters was identified by Nuclear Engineering and NRC, 3/21/07
PER 121776, SFSP item storage control, 3/17/07
PER 107546, Trend PER on posting discrepancies, 7/26/06
PER 99818, Unanticipated dose rate alarms received in the Unit 3 Moisture Separator Room, 3/23/06

Audits and Self-Assessments
Radiation Protection Integrated Trend Review Second Quarter FY 2006
Radiation Protection Integrated Trend Review Third Quarter FY 2006
Radiation Protection Integrated Trend Review Fourth Quarter FY 2006
BFN-RP-06-002, RP Surveys and Monitoring Self Assessment Report, 9/18/06 - 9/21/06
BFN-RP-06-007, Radiological Hazard Tags Snapshot Self-Assessment Report, 6/21/06

Records and Data Reviewed
Radiological Work Permit (RWP) No. 07272256, U2C14 Rx Bldg 2A RWCU HX Encapsulation (LHRA Various Dress), Rev. 0

Radiological Work Permit (RWP) No. 07272257, U2C14 Rx Bldg 2A RWCU HX Encapsulation (LHRA Resp, Various Dress), Rev. 0
Radiological Work Permit (RWP) No. 07272245, U2C14 Rx Bldg Replace 2C RHR Floating Head (High Rad, Various Dress), Rev. 0
Radiological Work Permit (RWP) No. 07272247, U2C14 Rx Bldg Replace 2C RHR Floating Head (High Rad, Respirator, Various Dress), Rev. 0
Radiological Work Permit (RWP) No. 07272247, U2C14 Rx Bldg Replace 2C RHR Floating Head (High Rad, Tyvek Hood, Various Dress), Rev. 0
Radiological Work Permit (RWP) No. 07280005, U2C14 DW Misc Maintenance (High Rad Various Dress), Rev. 0
BFNP Radiological Survey No. 030107-11, Unit 2 Drywell 559' Rod Gallery, 3/1/07
BFNP Radiological Survey No. 031707-18, Unit 2 DW 579' RHR Chem Decon Survey, 3/17/07
BFNP Radiological Survey No. 031007-14, Unit 2 Drywell 584', 3/10/07
BFNP Radiological Survey No. 031907-01, Unit 2 Drywell 584', 3/19/07
BFNP Radiological Survey No. 022607-05, Unit 2 RXB 565' General Area, 2/26/07
BFNP Radiological Survey No. 021907-01, Unit 3 RXB 565' General Area, 2/19/07
BFNP Radiological Survey No. 022707-10, Unit 2 RXB 565' A&C RHR Hx Room, 2/27/07
BFNP Radiological Survey No. 032306-18, Unit 3 TB 586' Moisture Separator Room, 3/23/06

Procedures, Manuals, and Guidance Documents
Operations Section Instruction Letter (OSIL)-16, Operations Keys and Equipment, Attachments 1 - 3, 7/31/06
OSIL-103, Key Control and Accountability, 4/17/1999
SPP (Standard Programs and Processes)-3.1, Corrective Action Program, Revision (Rev.) 11B1.
SPP-5.1, Radiological Controls, Rev. 5
Radcon Department Procedure (RCDP)-1, Conduct of Radiological Controls, Rev. 2
RCDP-3, Administration of Radiation Work Permits (RWPs), Rev. 2
RCDP-7, Bioassay and Internal Dose Program, Rev. 0
RCI (Radiological Control Instructions)-1.1, Radiation and Contamination Surveys, Rev. 122
RCI-8.1, Internal Dosimetry Program Implementation, RCI-8.1
RCI-9.1, Radiation Work Permits, Rev. 50
RCI-15.5, Primary System Survey Procedure, Rev. 4
RCI-17, Control of High Radiation Areas and Very High Radiation Areas, Rev. 51

2OS2: ALARA Planning and Controls

Problem Evaluation Reports (PERs)
PER 119881 The radwaste group exceeded the dose estimate for the work week by 106.36%. Goal was 220 mrem and actual was 454 mrem. The extra dose was a direct result of Unit 2 preoutage work and the unanticipated scram of Unit 3.
PER 120464 Error in work package resulted in boilermakers receiving an additional 100 mrem due to having to disconnect piping a second time.
PER 120548 The chemical decontamination of the “2A” and “2C” RHR Hex was not as successful as it could have been.
PER 121064 On 3/7/07 it was recognized that work was progressing on elevation 584 without the benefit of contingency temporary shielding.

PER 121186 Worker used wrong RWP
PER 121336 ALARA Planning Report had to be revised due to low man-hour estimate for Operations personnel performing rounds, inspection activities, system manipulations and other activities that were not accounted for in the outage base schedule.
PER 121347 Worker used wrong RWP
PER 121372 RWP intended for EPU work being used for other activities. Of 2447 mrem attributed to work step only 131 mrem attributed to EPU work.
PER 121510 RWP intended for low risk activities being used for other than intended work.
PER 121690 During the U2C13 control rod drives exchanged had average contact dose rate of 4.35 rem/hr. During U2C14 the average was 16.541 rem/hr or an increase of 380%.

Audits and Self-Assessments
Browns Ferry Nuclear Fiscal Year 2006 Annual ALARA Report
U3C12 Outage ALARA Report
Browns Ferry Nuclear Power Plant Long-Term Collective Radiation Exposure Reduction Plan 2007 - 2011

Records and Data Reviewed
Spreadsheet: BFN Source Term / Dose Reduction Initiatives
Graph: BFN Historic Trends and Current Status of Plant Source Term
Listing: Top 15 ALARA Planning Reports for Dose
ALARA / Radwaste Committee Meeting Minutes 4/25/06, 5/12/06, 6/13/06, 8/8/06, 10/9/06 and 1/12/07.
ALARA Planning Report 06-41, U3C12 Outage- Refuel Floor Activities
ALARA Planning Report 06-42, U3C12 Outage- Scaffolding Support
ALARA Planning Report 06-47, U3C12 Outage- Under-Vessel Work Activities

ALARA Planning Report 06-62, U3C12 Outage- Weld Repair 3B RWCU Regenerative Heat Exchanger with contingency to repair 3C RWCU Regenerative Heat Exchanger
ALARA Planning Report 06-72, U3C12 Outage- Insulation & Shielding Remove, Maintain and Install
ALARA Planning Report 07-08, U2C14 Outage- Replace 2 “C” RHR Heat Exchanger Floating Head and Associated Work Activities
ALARA Planning Report 07-37, U2C14 Outage- Recirc Pump Seal/ Impeller
ALARA Planning Report 07-57, U2C14 Outage- Repair, Maintenance, Testing SI’s on Various Systems Within the Turbine Building
ALARA Planning Report 07-44, U2C14 Outage- Radiation Protection Support
ALARA Planning Report 07-50, U2C14 Outage- Operations Support
ALARA Planning Report 07-55, U2C14 Outage- Chemical Decontamination Project for Recirc, RWCU and RHR Systems

Procedures, Manuals, and Guidance Documents
RCI-1.1, Radiation Operations Program Implementation, Rev. 122
RCI- 9.1, Radiation Work Permits, Rev.50
RCI-15.1, Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable (ALARA), Rev 34
RCI- 15.3, ALARA/ Radwaste Committee, Rev.21

RCI- 25, Closed Circuit Television/ Audio System, Rev. 4
RCI- 26, Radiation Protection Standards and Expectations, Rev.7
RCI- 27, Source Term Reduction and Control, Rev. 3
SPP-5.1, Radiological Controls, Rev.5
SPP- 5.2, ALARA Program, Rev.3
SPP- 7.0, Work Management, Rev.1
SPP- 7.1, On Line Work Management, Rev.8
SPP- 7.2, Outage Management, Rev.8

2PS1: Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

Records, Data, and Reports Reviewed
BFNP Investigation of Tritium Releases to Groundwater, June 2006
Trend Graph for Groundwater Monitoring Results (Wells BH-25, BH-38B, CT, Dewat A, Dewat E, R1, R2, and R3), printed 3/20/07
Memo to File: BFNP Review of Survey Results from 10CFR50.75(g) Questionnaire and File Search, 8/28/06
TVAN Record Entries for Decommissioning Pursuant to 10 CFR50.75(g), various dates
PER 96105, 10CFR50.75(g) Site Contamination Records Kept by Utilities for Decommissioning, 1/26/06

2PS2 Radioactive Material Processing and Transportation

Procedures, Manuals, and Guides
Browns Ferry Nuclear Plant Process Control Program Manual (PCP), Rev. 3
RCDP-1, Conduct of Radiological Controls, Rev. 2
RCI-1.1, Field Operations Program Implementation, Rev. 116
RCI-7, Receipt of Radioactive Materials, Rev. 16
RCI-15.3, ALARA/Radwaste Committee, Rev. 16
RWTP (Rad Waste Technical Procedure) -100, Radioactive Material/ Waste Shipments, Rev. 0004
RWTP-101, 10 CFR 61 Characterization, Rev. 0
RWTP-102, Use of Casks, Rev.1
SPP 3.1, Corrective Action Program, Rev. 11
SPP-5.1, Radiological Controls, Rev. 5

Shipping Records and Radwaste Data
10 CFR 61 Analysis Report, dry active waste (DAW) (Unit 1) Smears, Dated 09/28/05
10 CFR 61 Analysis Report, DAW (Unit 0, 2 & 3) Smears, Dated 09/28/05
10 CFR 61 Analysis Report, CWPS Resin, 09/28/05
Browns Ferry Nuclear Plant - Personnel Qualified to Ship Radioactive Material/Waste Letter to File, Dated July 20, 2005
Shipment No. 050423, Contaminated Equipment Shipped to Framatome, 04/20/05
Shipment No. 050424, Contaminated Equipment Shipped to Framatome, 04/20/05
Shipment No. 050710, Unit 1 Shrouds Bolts Shipped to Duratek, 07/14/05
Shipment No. 051015, Irradiated Control Rod Blades Shipped to Barnwell, 10/25/05
Shipment No. 051128, Unit 1 Control Rod Drives Shipped to Duratek, 11/17/05

Shipment No. 060317, Unit 3 MSRV Cartidges/Valves Shipped to Wyle, 03/10/06
Shipment No. 061104, Resin Sample Shipped to Teledyne Brown, 11/07/06
Shipment No. 070340, Dewatered Condensate Resin Shipped to Duratek, 03/21/07

Corrective Actions Program Documents
Nuclear Assurance - TVAN-Wide - Audit Report No. SSA0502 - Radiological Protection and Control Audit, Dated January 19, 2006
PER 81364, Radioactive Material Shipments to Framatome Reported Above DOT Shipping Limits, Dated 04/26/05
PER 99993, Framatome Box No. 003 Containing CRD Handling Equipment was Packaged for Shipment with the Smearable Contamination Levels too High to Allow Shipment as SCO, Dated 03/31/06
PER 109364, A Radioactive Material Shipment Survey was not Performed Prior to Closure of Westinghouse Refuel Equipment Boxes, Dated 08/31/06
Snapshot Self-Assessment Report, Assessment No. BFN-RP-06-008, Dated August 17, 2006

4OA1: Performance Indicator Verification

Procedures
SPP-3.4, Performance Indicator and MOR Submittal, Rev. 5
Database Print-out of ED Dose-Rate Alarms Greater than 1000 mrem/hr, 3/2/07
CI-138, Reporting NEI Indicators, Rev.3

Documents/Records
PER 94995, Unanticipated dose rate alarms in 3B SJAE, 1/09/06
PER 106829, PERs for dose rate alarms, 7/14/06
PER 121615, LHRA Door 499 A, 3/14/07
BFNP Radiological Survey No. 01906-16, Unit 3 TB 586' SJAE Room, 1/09/06
BFNP Radiological Survey No. 01906-17, Unit 3 TB 586' SJAE Room, 1/09/06
0-SI-4.8.A.5-1, Appendix I Dose Calculations-Liquid Effluents, Rev. 16, Performed 2/15/07
0-SI-4.8.B.3, Appendix I Dose Calculations-Airborne Effluents Rev.22, Performed 1/16/07
0-SI-4.8.B.3, Appendix I Dose Calculations-Airborne Effluents Rev.22, Performed 2/14/07
0-SI-4.8.B.3, Appendix I Dose Calculations-Airborne Effluents Rev.22, Performed 3/14/07