UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

May 5, 2003

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 50-259/2003-02 , 50-260/2003-02 , 50-296/2003-02

Dear Mr. Scalice:

On April 5, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry 1, 2, and 3 reactor facilities. The enclosed integrated inspection report documents the inspection results, which were discussed on April 23, 2003 with Mr. Ashok Bhatnager 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.

This report documents one self-revealing finding of very low safety significance (Green), which was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. In addition, a licensee identified violation is listed in section 4OA7 of this report. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Browns Ferry.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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/

Stephen J. Cahill
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: NRC Integrated Inspection Report 50-259/03-02, 50-260/03-02, 50-296/03-02
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-259, 50-260, 50-296
License Nos:	DPR-33, DPR-52, DPR-68
Report No:	50-259/03-02, 50-260/03-02, 50-296/03-02
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, & 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	December 29, 2002 - April 5, 2003
Inspectors:	B. Holbrook, Senior Resident Inspector W. Bearden, Senior Resident Inspector, Unit 1 (Section1R08) E. Christnot, Resident Inspector R. Carrion, Senior Project Engineer (Section 1R06) D. Jones, Senior Health Physics Inspector (Section 2S01, 4OA1) R. Hamilton, Health Physics Inspector (Section 2S02) A. Nielsen, Health Physics Inspector (Section 2PS2)
Approved by:	Stephen J. Cahill, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259/2003-002, 05000260/2003-002, 05000296/2003-002 ; Tennessee Valley Authority; on 12/29/2002-4/4/2003; Browns Ferry Nuclear Plant, Units 1, 2 and 3; Event Follow-up.

The report covered a three-month period of inspection by resident inspectors and an announced inspection by a regional senior project engineer, and three health physicists. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating System

- Green. An inadequate work control authorization process procedure and poor trouble shooting techniques resulted in the failure of Unit 3 High Pressure Coolant Injection System (HPCI). The procedure did not provide specific procedural direction for assessing all possible impacts of the proposed work activity on the associated plant system.

  A self revealing non-cited violation of Technical Specification 5.4.1a was identified. This finding is greater than minor because the loss of the HPCI system had an actual impact on safety in that an important mitigating system was not available to respond to postulated events. The finding is of very low safety significance because all other mitigating systems were available to provide core cooling injection (Section 4OA5.).

B. Licensee Identified Findings

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

Report Details

Summary of Plant Status

Unit 1 has been shut down since March 19, 1985, and has remained in a long-term lay-up condition with the reactor defueled. The licensee has initiated long-term recovery actions to return the unit to an operational status.

At the beginning of the report period Unit 2 power was at 88% Rated Thermal Power (RTP) due to end-of-cycle coastdown. The unit operated at maximum coastdown thermal power with the following exceptions: On January 26, power was reduced to 66% RTP due to the closing of the #3 and #4 main turbine control valves following the failure of an electronic control card. Unit power was returned to near maximum power on January 28. The unit commenced the scheduled refueling outage on February 24. On March 15, the unit was taken critical, ending the refueling outage. On March 20, the unit was at 80% RTP when, due to a stuck closed feedwater heater outlet valve, the unit was shut down. On March 24, the unit was taken critical and 100% RTP was achieved on March 25. On March 26, the unit was manually scrammed following a trip to both reactor recirculating pumps. The unit was taken critical on March 27, reached 100% RTP on March 31, and operated at or near 100% RTP during the remainder of the inspection period.

Unit 3 operated at or about 100% RTP with the following exceptions: On February 14, power was reduced to 50% RTP to conduct planned maintenance and surveillance testing and to perform power suppression testing due to leaking fuel. The unit was returned to 100% RTP on February 24. The unit was returned to 100 % RTP on April 2. The unit operated at or near 100% RTP during the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (Partial and Complete Walkdown)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed a partial walkdown of three safety systems listed below to verify redundant or diverse train operability, as required by the plant Technical Specifications (TS). In some cases, the system was selected because it would have been considered an unacceptable combination from a Probabilistic Safety Assessment (PSA) perspective for the equipment to be removed from service while another train or system was out of service. The inspectors' walkdown was to verify that selected breaker, valve position, and support equipment were in the correct position for support system operation. The walkdown was also to identify any discrepancies that impact the function of the system that could lead to increased risk. Also, the inspectors reviewed that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the availability and functional capability of mitigating systems or barriers. The inspectors' observations of equipment and component alignment for the partial walkdowns were compared to the alignment specified in: Procedure 2-SR-3.5.1.2, Monthly Residual Heat Removal Valve Lineup Verification Loop 1, 2-OI-74, Residual Heat Removal, (Attachment 1, Valve Lineup, Attachment 2, Panel Lineup, and Attachment 3, Electrical Lineup); Core Spray (CS) electrical lineup checklist, CS panel lineup checklist and drawing 2-47E814-1, CS Flow.

- Unit 2 Residual Heat Removal (RHR) system Loop I Low Pressure Coolant Injection (LPCI) function during RHR Loop II planned maintenance
- Unit 2 core spray loop II during core spray loop I planned maintenance
- Unit 3 core spray division I during testing of Division II

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors reviewed Section 8.5 of the Updated Final Safety Analysis Report (UFSAR), licensee procedure 0-OI-57A, Switchyard and 4160 volt AC Electrical System, and Attachment 2, Switchyard and 4160 volt AC Electrical System Panel Lineup Checklist to conduct a detailed walkdown of the Unit 2 Emergency Diesel Generator/AC Power system. Inspectors conducted a complete review of system and components in the plant and main control room to verify proper breaker and switch alignment and that components were labeled and available for operation. The inspectors reviewed breaker and emergency bus and board lockout devices to verify they were reset and would not prevent operation of the electrical components. The inspectors reviewed emergency bus and board voltage readings to verify that the supplied voltage met TS and procedure requirements. The inspectors reviewed a sampling of Problem Evaluation Reports (PERs) to verify that deficiencies were being identified and corrected. The inspectors also reviewed system health reports, the maintenance rule report, and the operator workaround list to assess overall system condition to verify that identified deficiencies would not affect system function.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (Walkdown and Drill Observation)

a. Inspection Scope

The inspectors reviewed licensee procedure, SPP-10.10, Control of Transient Combustibles, and SPP-10.9, Control of Fire Protection Impairments and conducted a walkdown of the six fire areas listed below in order to verify a selected sample of the following: licensee control of transient combustibles and ignition sources; the material condition of fire equipment and fire barriers; operational lineup; and operational condition of selected components. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with the procedure SPP-10.9. In addition, the inspectors reviewed the Site Fire Hazards Analysis and applicable Pre-fire Plan drawings to verify the necessary fire fighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, were in place. The inspectors reviewed a sampling of fire protection related PERs to verify the licensee was identifying and correcting fire protection problems. Pre-fire Plan drawings and documents reviewed are included in the attachment to the report.

- Fire Area 9 (Units 1, 2 - 4kV shutdown board room C)
- Fire Area 14 (Unit 3 - 480V shutdown board room A)
- Fire Area 16 (Unit 2 and Unit 3 control building elevation 593)
- Fire Area 18 (Unit 2 battery and battery board room)
- Fire Area 20 (Unit 1 & 2 diesel generator rooms)
- Fire Area 21 (Unit 3 diesel generator rooms)

Fire Drill: The inspectors also observed an announced Quarterly fire drill at the intake structure conducted on backshift on February 20. The inspectors assessed fire alarm effectiveness, time required to notify and assemble the fire brigade, the selection and placement of equipment, and fire fighting strategies. The inspectors also observed the number of responders, communication techniques, and teamwork. The inspectors attended a post drill critique to assess licensee actions to review fire brigade performance and to identify areas for improvement. The inspectors compared their observations to the requirements in the licensee’s Fire Protection Report Volume 2, Section III, Fire Brigade Training.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal and external flooding events. The inspectors reviewed flood analysis documents including: UFSAR Section 2.4, Hydrology, Water Quality, and Marine Biology, which included Appendix 2.4A, Maximum Possible Flood; Design Criteria BFN-50-C-7105, Revision 7, Section 7.0, Internal Flooding Design Basis; Emergency Operating Instruction 3, Secondary Containment Control, Revision 9; Browns Ferry Unit 2 Individual Plant Examination, Appendix E, Generic Issues, Section E.1, Browns Ferry Internal Floods Analysis; and Browns Ferry Nuclear Plant Probabilistic Safety Assessment Initiating Event Notebook, Revision 0, Section 3, Initiating Event Frequencies for licensee commitments. The inspectors also interviewed cognizant licensee personnel knowledgeable about site flood protection measures and plant drainage plans.

External Flood Features: for external flooding protection features, the inspectors performed walkdowns of risk-significant areas, susceptible systems and equipment including the RHR Service Water Pump rooms, the Intake Structure Cable Tunnel, and the 500-kV Switchyard Cable Tunnels.

Internal Flood Features: for internal flooding protection features, the inspectors performed walkdowns of the Units 2 and 3 Reactor Buildings, the Unit 3 Diesel Generator Rooms "A" and "B," and the Units 2 and 3 Turbine Building Condensate Pits for flood-significant features such as level switches, room sumps, and door seals. Plant procedures for coping with flooding events were also reviewed to verify that licensee actions were consistent with the plant's design basis assumptions. The reviewed procedures are listed in the attachment.

The inspectors also reviewed the licensee's corrective action documents with respect to flood-related items identified in PERs written in 2002 and early 2003 to verify the adequacy of the corrective actions. The inspectors reviewed selected completed preventive maintenance procedures and work orders for selected level switches and pumps for completeness and frequency.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors conducted a review of the licensee's heat exchanger performance program, which consisted of periodically disassembling the safety related heat exchangers, cleaning, inspecting, eddy current testing heat exchanger tubes, plugging defective tubes, and reassembling the heat exchangers, to verify the requirements in licensing bases documents are met. The inspectors reviewed procedures N-ET-6, Eddy Current Testing, OMNI 892157 and OMNI 892161, to ensure that the RHR system heat exchangers 1B and 2C would be able to supply the necessary cooling as described in the UFSAR, Sections 4.8 and 6.4.4. The inspection focused on deficiencies that could mask degraded performance of the heat exchangers and/or result in common cause heat exchanger performance problems. Also assessed was whether the licensee had adequately identified and resolved heat sink performance problems that could affect multiple heat exchangers in mitigating systems.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

.1 Unit 2 Refueling Outage ISI Activities

a. Inspection Scope

The inspectors observed in-process ISI work activities and reviewed selected ISI records. The observations and records were compared to the TS and the applicable Code (ASME Boiler and Pressure Vessel Code, Sections V and XI, 1995 Edition, with 1996 Addenda) to verify compliance.

Calibration of Ultrasonic (UT) examination equipment and portions of the ongoing UT examination of the reactor vessel head vent six inch nozzle N6A weld were observed. The inspectors reviewed NDE examination reports for completed dry powder Magnetic Particle (MT), Liquid Penetrant (PT) and UT examinations. In addition, the inspectors reviewed the weld examination reports and radiographs of completed weld repairs. The reviewed reports are listed in the attachment to the report.

Qualification and certification records for examiners, equipment and consumables, and nondestructive examination procedures for the above ISI examination activities were reviewed. One Notice of Indication, associated with minor piping support deficiencies was reviewed by the inspectors. In addition, two PERs associated with ISI activities which had been documented in the licensee's corrective action program were reviewed.

b. Findings

No findings of significance were identified.

.2 Unit 1 Piping UT Examination Activities.

a. Inspection Scope

The inspectors reviewed selected records for Ultrasonic Examination (UT) activities performed on Main Steam (MS) and Feedwater (FW) piping in the Unit 1 Drywell and MS piping in the Unit 1 steam tunnel and turbine building . The licensee had performed extensive UT exams to determine the effects of long term lay-up and to verify that piping wall thickness measurements satisfied design requirements. Various Unit 1 piping such as used in the recirc and core spray systems were manufactured from stainless steel and susceptible to stress corrosion cracking (SCC). Much of the piping in those systems has been removed and will be replaced as part of the Unit 1 recovery effort. However, based on previous experience with Unit 2 and Unit 3, significant portions of the MS and FW systems in the drywell and portions of the piping in these systems located in the turbine building remained in good condition and did not require replacement. Piping in these systems are manufactured from low alloy steel and not susceptible to SCC. The licensee performed the UT exams to verify their assumptions about the condition of the piping for those two systems to remain in use. The records documenting manual UT examination of piping segments are listed in the attachment.

The results of the UT examinations were compared to the applicable drawings and system design requirements to verify acceptable piping wall thickness readings. Qualification and certification records for examiners, equipment and procedures for the UT examination activities were reviewed.

b. Findings and Observations

Although examples of flow assisted corrosion (FAC) and localized corrosion were found during the UT examinations no flaw like indications or wall thickness values below acceptable were identified. The licensee concluded that the condition of the Unit 1 MS and FW piping was acceptable and comparable to the condition previously found on Unit 2 and Unit 3 MS and FW piping.

No findings of significance were identified.

1R11 Licensed Operator Requalification

Resident Inspector Quarterly Review of Testing and/or Training Activities

a. Inspection Scope

The inspectors observed portions of an operations crew performance during Simulator Evaluation Guide, OLP173S001, Reactor Startup and Heatup, Startup Instrumentation Failures, CRD Pump Trip, Control Rod Drift, High Worth Control Rod, LOCA, and Rod Drop (Contains SOER 90-3 and SER 6-00). The inspectors reviewed licensee procedures TRN-11.4, Continuing Training For Licensed Personnel, TRN-11.9, Simulator Exercise Guide Development and Revision, and OPDP-1, Conduct Of Operations, to verify the conduct of training; that the exercises contained high risk operator actions, the formality of communication, procedure usage, alarm response, control board manipulations, and supervisory oversight were in accordance with the above procedures.

The inspectors also reviewed previously identified deficiencies to verify that they were included in the current training. The inspectors attended the post exercise critiques to verify the licensee identified issues were comparable to issues identified by the inspectors.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two items listed below for the following: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule; (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); and (8) appropriateness of performance criteria for Systems, Structures and Components (SSCs)/functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions 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 -10 CFR 50.65, Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting -10 CFR 50.65, and SPP 3.1, Corrective Action Program.

- PER 01-005022-000, Unit 2 and 3 Backup Control (BUC) system has had multiple component failures within the last year. The system provides mitigating equipment outside the control room to shutdown and maintain the reactor shutdown. The inspectors reviewed the performance history of the BUC systems, especially for multiple component failures for the wide range reactor water level indicating instruments, 2-3-46B and 3-3-46B.

  The inspectors reviewed licensee corrective actions to support long term operational readiness such as re-engineering the electronic cards to use updated supportable components, such as capacitors; maintaining a like-for-like electronic card replacement; using a re-engineered card as a pattern to fabricated updated cards; and using the updated electronic cards as replacement parts.

- PER 02-12362 and 02-003226: Unit 2: Drywell head seal exceeded acceptance criteria for three consecutive as-found leak rate tests. The inspectors reviewed the as-found leak rate for the drywell head and compared the leak rate of 68 standard cubic feet per hour (scfh) to the administrative acceptance criteria of 1.8 scfh. The inspectors reviewed licensee immediate and long term corrective actions. The inspectors reviewed licensee corrective maintenance actions such as: clean the O-ring groove surfaces; remove water in the area of the drywell flange; eliminate water spray into the drywell; and to minimize washdown water into the drywell after the drywell head was set. The inspectors compared licensee actions to the requirements of SPP 3.1, Corrective Action Program. The inspectors reviewed the as-left leak rate test to verify that the test met the requirements of procedure 2-SI-4.7.A.g-2/FHA, Primary Containment Local Leak Rate Test: Flanges and Hatches.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For the eight emergent work and equipment issues listed below, the inspectors reviewed licensee actions taken to plan and control the emergent work activities to effectively manage and minimize risk. The inspectors verified that risk assessments were being performed as required by 10 CFR 50.65(a)(4). The inspectors reviewed: licensee procedure SPP-6.1, Work Order Process Initiation; SPP-7.1, Work Control Process; and 0-TI-367, BFN Dual Unit Maintenance to verify procedure steps and required actions were met. Also, the inspectors evaluated the adequacy of the licensee's risk assessments and the implementation of compensatory measures.

- Unit 2, average power range monitor #1 failure on January 12, work order (WO) 03-000516-000 and WO 02-008422-000 (emergent)
- Unit 2, loss of pressure switch 2-PS-64-67B, drywell pressure, and post accident monitoring instrument pressure indicator 2-PI-64-67B, drywell pressure, following an unanticipated loss of power when fuses were pulled as part of the Unit 2 HPCI maintenance outage, PER-03-000576-000 (emergent)
- Unit 2, troubleshoot and repair electrical ground on local indicator for the control circuitry of diesel generator A, WO 030301800 (emergent)
- Unit 2, Reactor Core Isolation Cooling system out of service to repair pressure control valve 71-22 sensing line union leak (emergent)
- Unit 3, troubleshoot and repair a hard, low impedance, ground on phase A of the 480V shutdown board 3B, WO 03-000963-000 which required that power to systems important to safety be momentarily turned off and back on (emergent)
- Unit 3, the A inlet damper for diesel generator ventilation system failed to open during surveillance testing of diesel generator 3B, WO 030302800 (emergent)
- All Units: during removal of the residual heat removal service water pump C2 the discharge valve 2-FCV-23-40 failed to close completely, PER-003073-000 (emergent)

- All Units: the automatic tap changers, for the unit station service and common stations service transformers, may not have sufficient voltage to correctly operate PER-003003-000 (emergent)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed operator performance and actions during the following non-routine conditions to verify performance was in accordance with licensee procedures and regulatory requirements. The inspectors reviewed operator logs, computer data, and control room chart data to verify operator actions were appropriate for plant conditions and as required by operator training. Inspector observations were compared to licensee procedure OPDP-1, Conduct Of Operations, SPP- 3.5, Regulatory Reporting Requirements, and 10 CFR 50.72 and 10 CFR 50.73, Event Reporting Guidelines. The non-routine events reviewed included the following:

- On February 16, following the Unit 2 refueling outage, the HPCI system tripped during testing. The Unit was in MODE 1, when the HPCI system tripped and isolated on high steam flow shortly after it was started. The cause of the trip was a loss of the speed signal to the HPCI turbine control network. This loss of signal allowed the turbine control valve to continuously travel in the open direction until the trip.

- On February 24, the inspectors observed and reviewed operator response to a Unit 2 reactor scram. The inspectors assessed operator performance leading up to the scram; the use of abnormal and emergency procedures to restore and control reactor water level; and verify plant systems automatically responded to the scram signal as expected. The inspectors also reviewed licensee actions to determine 10 CFR 50.72 and 10 CFR 50.73 reportability. The inspectors reviewed PER 03-003081-000, BFN Unit 2 Scram on Low Reactor Water Level, to verify the requirements of procedure SPP-3.1, Corrective Action Program, were met. Procedures used to assess operator performance are included in the attachment. Additional information is documented in section 4OA3.2 of the report.

- On March 26, the inspectors observed operator activities, monitored licensee controls over recovery activities, and reviewed selected items following a Unit 2 manual scram, to verify that procedure and regulatory requirements were met. The scram was initiated following a trip of a second recirculating pump. The first pump had tripped about two hours prior to the second pump trip.

The inspectors compared their reviews and observations to 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-GOI-100-1A, Unit Startup and Power Operation to verify procedure requirements were met. Additional information is documented in Section 4OA3.7 of the report.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

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

- Diesel Generator 3C following the failure of Fan A to restart within the time frame specified in the test procedure (PER 03-000514-000)

- Unit 2 rod block monitor, 2-RBM-92-5B, loss of communication with average power range monitor, 2-APRM-92-1, due to a failed fiber direct data interface card (PER 03-000531-000)

- RHR Service Water Pump B1 due to low flow during performance of surveillance instruction (PER 03-001925-000)

- RWCU regenerative heat exchangers maximum number of thermal cycles allowed by paragraph N-415.1 of Section III of the ASME Code, 100 during the life of the plant, not being tracked (PER 03-001802-000)

- Unit 2 and 3 qualification of Rockbestos electrical cables, with one installed in Unit 3 drywell and 211 installed outside the drywells of Units 2 and 3, due to a possible alternate resin being used in the insulation material (PER 03-002067-000)

- Unit 3 Core Spray Loop II when a discharge pressure instrument was found inoperable (PER 03-002341-000)

b. Findings

No findings of significance were identified.

1R16 Operator Work-Around (OWA) Review and Cumulative Affect Assessment

a. Inspection Scope

The inspectors reviewed the status of OWAs for Units 2 and 3 to determine if the functional capability of the system or operator reliability in responding to an initiating event was affected. The review was to evaluate the effect of the OWA on the operator's ability to implement abnormal or emergency operating procedures during transient or event conditions. The inspectors conducted a detailed review of Unit 2 OWAs to verify that items scheduled for repair during the recent refueling outage were corrected. The inspectors also completed an assessment for the overall cumulative affect of operator ability to perform actions in response to events. The inspectors compared their observations and licensee actions to the requirements of Operations Directive Manual 4.11, Operator Work Around Program and TVAN Standard Department Procedure OPDP-1, Conduct of Operations. The inspectors reviewed procedure 3-SR-3.4.4.1, Manual Calculation of Unidentified, Identified, and Total Leakage completed daily for the week of March 21, to verify operator actions identified for OWA -2003-0018 were completed. The two OWAs reviewed were:

- Unit 3: 3-077-OWA-2002-0105, Drywell equipment drain sump integrator creeping requiring manual calculations of leakage (WO 02-005589)
- Unit 2: 2-073-OWA-2003-0018, HPCI 2-LS-73-8A, level switch spuriously brings in the exhaust drain pot level high alarm during turbine operations requiring operators to respond to the alarm (WO 02-002840-000)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modification - Annual Review

a. Inspection Scope

The inspectors reviewed the permanent plant modification to replace the existing Unit 2 reactor manual control system timer with a new computer based programmable timer: Design Change Notice (DCN) 50942, “Implement stage 1 of DCN to replace the reactor manual control system timer with a PLC based system” (WO 02-002892-000). The inspectors reviewed criteria in licensee procedures SPP-9.3, Plant Modifications and Engineering Change Control, SPP-7.1, Work Control Process, SPP-8.3, Post-modification Testing, and SPP-8.1, Conduct of Testing, to verify the risk significant plant modification, was developed, reviewed, and approved per the procedure requirements. The inspectors observed field work to establish working conditions in safety related panels, connection of jumpers, test leads, removal of the old timer, and installation of the new timer and a spare timer. The inspectors reviewed the post-modification work and test package following the DCN implementation with the unit online to verify the design basis, licensing bases, and TS required performance for the system had not been degraded as a result of the modification. The inspectors also reviewed the same modification work package developed to replace the existing Unit 3 timers in February 2003, with the unit on-line.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors evaluated the following six activities by observing testing and/or reviewing completed documentation to verify that the PMT was adequate to ensure system operability and functional capability following completion of associated work. The inspectors reviewed licensee procedure SPP-6.3, Post-Maintenance Testing, to verify that testing was conducted in accordance with procedure requirements. For some testing, portions of MMDP-1, Maintenance Management System, were referenced.

- Unit 2, PMT on core spray pump motor 2A following maintenance per WO 02-125580-000
- Unit 2, PMT on reactor manual control timer per test number 2-PMT-BF-085-031
- Unit 3, PMT on temperature switch 3-TS-001-0029C, following maintenance per test procedure 3-SR-3.3.6.1.2(1D/A2)
- Unit 2, PMT on diesel generator B following maintenance per test procedure 0-SR-3.8.1.9 (B), Emergency Load Acceptance Test
- Unit 2, PMT on control rods per technical instruction 0-TI-20, Control Rod Drive System Testing and Troubleshooting
- Unit 2, PMT on the variable frequency drive system per instruction PMT-2-PMT-50869

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Unit 2 Refueling Outage

a. Inspection Scope

Risk

Prior to the outage scheduled for February 24, - March 27, the inspectors reviewed the Unit 2 Cycle 12 Outage Risk Assessment Report, to verify the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. The inspectors' review was compared to the requirements in licensee procedure SPP-7.2, Outage Management. The review was also to verify that, for identified high risk significant conditions, due to equipment availability and/or system configurations, contingency measures were identified and incorporated into the overall outage and response plan. The inspectors frequently discussed posted risk conditions with operations and outage personnel to assess licensee personnel knowledge of the risk condition and mitigation strategies.

Shutdown and Cooldown Process

The inspectors observed selected activities and monitored licensee controls over outage activities listed below to verify that procedural and regulatory requirements were met. The inspectors compared their observations to licensee procedure SPP-12.1, Conduct Of Operations and 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations, to verify procedure requirements were met. Part of the activities observed included the following:

- Unit power reduction
- Core thermal limit verification
- Reactivity monitoring and control
- Startup, shutdown and realignment of components and systems
- Realignment and transfer of AC power sources
- TS instrument and system performance verification

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, 0-OI-72, Auxiliary Decay Heat Removal System and conducted a main control room panel and system walkdown to verify system availability. The inspectors conducted a review of the increased outage risk condition of Orange, for the removal of decay heat, to verify the plant conditions and systems identified in the risk mitigation strategy were available to remove decay heat.

The inspectors reviewed operational logs to verify procedure and TS requirements to monitor and record reactor coolant temperature were met. In addition, the inspectors reviewed the following two items:

- Controls implemented to ensure that outage work was not impacting the ability of operators to operate spent fuel pool cooling and RHR shutdown cooling
- Monitoring of decay heat removal process and spent fuel pool water temperature profile

<u>Reactivity Control</u>

The inspectors observed licensee performance during shutdown, outage, refueling, and startup activities to verify reactivity control was conducted in accordance with procedure and TS requirements. The inspectors conducted a review of outage activities and risk profile to verify activities that could cause reactivity control problems were identified. Inspector observations were compared to procedure SPP-10.4, Reactivity Management to verify procedure and TS requirements were met. Reactivity manipulations observed included the following:

- Power reduction with control rods and recirculation flow
- Fuel movement from the core to the spent fuel pool
- New fuel movement from the spent fuel pool to the core
- Control rod movement for testing
- Control rod withdrawal for criticality
- Control rod withdrawal for unit startup and power increase

Inspectors observed the following items to assess licensee performance in the respective area:

<u>Inventory Control</u>
- Reactor water inventories and controls including flow paths, system configurations, and alternate means for inventory addition
- Operator monitoring and control of reactor temperature and level profiles

<u>Electrical Power</u>
- Controls over electrical power systems and components to ensure emergency power was available as specified in the outage risk report
- Controls and monitoring of electrical power systems and components and work activities in the power transmission yard
- Operator monitoring of electrical power systems and outages to ensure that TS requirements were met
- Review of clearance activities to verify that equipment was identified and controlled to support work and testing activities and that equipment was correctly returned to service or standby conditions

Containment Control and Closure

- Confirm secondary containment requirements during fuel movement
- Verify leak rate and PCIV timing test results met TS requirements
- Verify torus and drywell walkdown and closeout prior to unit restart

Refueling Activities

- Fuel removal from the reactor core to the spent fuel pool
- Fuel sipping to identify leaking fuel
- Core alterations

Additional Procedures and documents reviewed are listed in the attachment of the report.

b. Findings

No findings of significance were identified.

.2 Unit 2 Forced Outage

a. Inspection Scope

The inspectors observed and monitored licensee activities to troubleshoot and complete repairs following the Unit 2 forced outage on March 20, due to a stuck closed feed water heater outlet valve. The inspectors observed selected operator and maintenance activities to monitor overall performance, procedure usage, command and control, and crew communications, to verify procedure requirements were met. The inspectors also observed operator performance to conduct TS required actions and parameter status checks to verify regulatory requirements were met. The inspectors compared their reviews and observations to 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-GOI-100-1A, Unit Startup and Power Operation to verify procedure requirements were met. Part of the activities observed and items reviewed included the following:

- Reactor shutdown and cool down to Mode 4
- Placing shutdown cooling in operation
- Shutdown risk assessment
- Reactor startup

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors either witnessed portions of surveillance tests or reviewed test data for the six risk-significant SSC's, listed below, to verify the tests met TS surveillance requirements, UFSAR commitments, and in-service testing (IST) and licensee procedure requirements. The inspectors' review was to confirm that the testing effectively demonstrated the SSCs were operationally capable of performing their intended safety functions. IST data was compared against the requirements of licensee procedures 0-TI-362, Inservice Testing of Pumps and Valves, and 0-TI-230, Vibration Monitoring and Diagnostics. The surveillances either witnessed or reviewed included:

- 2-SR-3.5.1.6, (RHR II), Quarterly RHR System Rated Flow Test Loop II
- 2-SR-3.1.3.2, Control Rod Exercise Test for Fully Withdrawn Control Rods
- 2-SR-3.5.2.6, (CS I), Core Spray Flow Rate Loop I
- 2-SI-4.2.B-60(I), Core Spray Area Cooler Fan Thermostat Calibration and Functional Test
- 3-SR-3.1.3.2, Control Rod Exercise Test for Fully Withdrawn Control Rods
- 3-SR-3.5.3.3, RCIC System Rated Flow at Norman Operating Pressure (IST)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed licensee procedures 0-TI-405, Plant Modifications and Design Change Control, 0-TI-410, Design Change Control, SPP-9.5, Temporary Alterations, and the two temporary modifications listed below to ensure that procedure and regulatory requirements were met. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify the modifications had not affected system operability/availability. The inspectors reviewed selected completed work activities and walked down portions of the systems to verify that installation was consistent with the modification documents and Temporary Alteration Control Form (TACF).

- TACF 2-03-002-078, disable air operator for valve 2-FCV-78-26, fuel pool cooling system, with valve in open position
- TACF 3-03-001-092, Unit 3 OPRM trip function bypass alteration

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed licensee performance in the control room simulator, technical support center and operations support center during an emergency training exercise on January 23. The drill was to focus on degraded plant conditions that led to implementation of the Severe Accident Management Guidelines. The inspectors review was to verify implementation of licensee procedures NP-REP, Radiological Emergency Plan, Browns Ferry Emergency Plan Implementing Procedures, SPP- 3.5, Regulatory Reporting Requirements, and OPDP-1, Conduct of Operations. The inspectors assessed operator performance, formality of communications, event classifications, and offsite emergency notifications to verify they were in accordance with the requirements of the above procedures. In addition, procedure usage, alarm response, control board manipulations, and supervisory oversight were evaluated to verify the procedure requirements were met. The inspectors also reviewed drill documents to verify that drill evaluations focused on improvement items identified during previous drills. The inspectors attended the post-exercise critiques to verify the licensee identified issues were comparable to issues identified by the inspectors.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Control To Radiologically Significant Areas

.1 Access Controls

a. Inspection Scope

Licensee program activities for monitoring workers and controlling their access to radiologically significant areas and tasks were evaluated. The inspectors assessed adequacy of procedural guidance; directly observed implementation of administrative and established physical controls; and assessed resultant worker exposures to radiation and radioactive material. Radiation worker and Health Physics Technician (HPT) proficiency in implementing Radiation Protection (RP) program activities were appraised.

Routine work activities in specific Radiological Controlled Area (RCA) locations were observed. The inspectors evaluated the adequacy of established physical and administrative controls including postings, barricades, procedural guidance, and radiation work permits for High Radiation Areas (HRAs) and Locked High Radiation Areas (LHRAs) through procedure reviews, direct observation of established controls, and interviews with workers. The inspectors performed independent confirmatory radiation surveys in accessible areas on the 565 foot (‘), 593', 621', and 639' elevations of the Unit 2 (U2) Reactor Building. The results of these surveys were compared to current licensee survey documentation. Electronic alarming dosimeter (EAD) set points were reviewed for consistency with expected work area dose rates. Radiation worker performance with respect to procedural guidance and HPT proficiency were assessed based on interviews and work observation. The workers knowledge of their expected response to an EAD dose or dose rate alarm was also assessed through interviews.

The inspectors reviewed procedural guidance for control of access to highly radioactive irradiated materials stored in spent fuel pool and discussed those controls with the Radiological Protection Supervisor and the Radiological Control Program Manager. Implementation of those controls was observed during tours of the Refuel Floor.

The inspectors evaluated the adequacy of licensee procedures for internal dose assessment and reviewed calculations for the two highest internal doses incurred during the U2 Refueling Outage (RFO).

License procedures and activities related to access controls were evaluated for consistency with Title 10 Code of Federal Regulations (10 CFR) 19.12; 10 CFR 20; and TS Section 5.4, Procedures, and Section 5.7, High Radiation Areas. Licensee access control related procedures, reports and records reviewed during the inspection are listed in Section 2OS1 of the Attachment to this report.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Problem Identification and Resolution</u>

a. <u>Inspection Scope</u>

Issues identified through self-assessments and PER document associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with responsible licensee representatives. The inspectors assessed the licensee’s ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure TVAN Standard Programs And Processes, SPP-3.1, Corrective Action Program, Revision (Rev.) 4. Specific assessments, audits, and PER documents reviewed and evaluated in detail for this inspection area are identified in the Attachment to the report.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The plant collective exposure history for the years 2000 through 2003, based on the data reported pursuant to 10 CFR 20.2206 (c), was reviewed and discussed with the licensee. Implementation of the licensee's ALARA program during the U 2 RFO was observed and evaluated by the inspectors during the weeks of February 24-28 and March 10-14, 2003. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for the five outage work activities expected to incur the maximum collective exposures during the U2 RFO (U2C12) and five significant outage work activities for the most recent Unit 3 refueling outage (U3C10). Incorporation of the planning, established work controls, and expected dose and dose rates into ALARA pre-job briefings and Radiation Work Permits (RWPs) for those activities was also reviewed. Those elements of the ALARA program were evaluated for consistency with the methods, practices, and philosophy delineated in the licensee's Station ALARA Program. The inspectors also conducted independent confirmatory radiation surveys of selected job sites and general area surveys of five elevations in the U 2 reactor building to assess the accuracy of the dose rates recorded on survey maps for work areas in the U 2 Reactor Building.

Selected elements of the licensee's source term reduction and control program were examined. Program data, radiation field monitoring and trending, temporary shielding and shutdown and purification controls were examined by the inspectors to determine whether the program was effective in controlling and reducing exposure. Inspectors compared surveys from prior outages with the current outage to determine the expected radiological impact of having an abnormally high number of leaking fuel pins. Inspectors reviewed alterations in work sequencing necessitated by elevated dose-rates and observed the actions that were taken to compensate by altering schedules, flushing, dilution, shielding and system realignment.

Documentation of the licensees declared pregnant worker program was reviewed for six participants. This review included documentation of declaration of pregnancy or intent to become pregnant, pregnancy consultation interview checklist, release of prenatal radiation exposure information, exposure report printouts, and one withdrawal from declared pregnant worker program.

Through the above reviews and observations, the licensee's ALARA program implementation and practices were evaluated by the inspectors for consistency with 10 CFR Part 20 requirements and procedural guidance documented in Section 2OS2 of the Attachment to this report. Incurred exposures were evaluated for consistency with 10 CFR Part 20 dose limits, and licensee administrative procedures.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed an audit and self assessments related to the ALARA program. The audit and self assessments were evaluated against periodic program review requirements specified in 10 CFR 20. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with TVAN Standard Programs And Processes, SPP-3.1, Corrective Action Program, revision 4. Reviewed documents are listed in the Attachment to the report.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

.1 Waste Processing and Characterization

a. Inspection Scope

During the week of February 24, the inspectors evaluated licensee methods for processing and characterizing radioactive waste (radwaste). Inspection activities included direct observation of processing equipment for liquid and solid radwaste and evaluation of waste stream characterization data.

Liquid radwaste equipment was inspected for general condition and licensee staff were interviewed regarding system changes, component function, and equipment operability. Inspected equipment included components of the reverse osmosis system and phase separation hold-up tanks. In addition the inspectors evaluated radwaste areas for abandoned and out-of-service equipment.

The solid radwaste processing system components between the phase separation tanks and the shipping container fill-head were walked-down. System equipment was inspected for material condition and for configuration compliance with the UFSAR. A technician was interviewed to assess knowledge of sludge transfer processes and solid radwaste system operations. Procedural guidance involving transfer of sludge and filling of waste packages was evaluated. Reviewed documents are listed in the attachment to this report.

Licensee radionuclide characterizations for each major waste stream were evaluated. For Condensate Waste Phase Separator (CWPS) sludge, Reactor Water Cleanup Phase Separator (RWCU) sludge, and Dry Active Waste (DAW) the inspectors evaluated Process Control Program (PCP) requirements and licensee procedural guidance. Analyses for hard-to-detect nuclides were assessed and independent scaling factor calculations were performed and compared to licensee results. Radionuclide data were reviewed for the period 1999-2002 to determine the stability of plant waste streams. For selected shipment records, waste classification calculations were independently performed and compared to licensee results. The licensee's use of a combined U-2/U-3 DAW stream was assessed in terms of its impact on waste classification and shipment type determination.

Radwaste processing activities were reviewed for compliance with 10 CFR Part 50.59 and consistency with the licensee 's PCP and UFSAR Chapter 9. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 61.55 and guidance provided in the Branch Technical Position (BTP) on Radioactive Waste Classification.

b. Findings

No findings of significance were identified.

.2 Transportation

a. Inspection Scope

During the week of February 24, the inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included review of shipping related documents and direct observation of shipment preparation activities.

The inspectors assessed the adequacy of selected parts of two shipping-related procedures and one corporate level shipping guidance manual. Five shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The licensee's procedure for opening and closing their Type A shipping cask was compared to recommended vendor protocols. Training records for five individuals qualified to ship radioactive material were checked for completeness and the training curriculum provided to these workers was evaluated.

The inspectors directly observed the preparation of a contaminated laundry shipment and interviewed technicians regarding dose limits, vehicle placarding, and other shipping regulations. In addition, the inspectors observed a shipment of solidified sludge and noted shipping container markings and licensee techniques for radiation surveys. The inspectors called the 24-hour emergency response phone number to determine whether the operator had knowledge of the sludge shipment and would be able to provide prompt emergency response information in the event of an accident.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, 49 CFR Parts 170-189; as well as guidance provided in NUREG-1608, NUREG-1660, and applicable licensee procedures. Training activities were assessed against 49 CFR Part 172 Subpart H, and the guidance documented in NRC Bulletin 79-19. Documents reviewed during the inspection are listed in of the Attachment to the report.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

Selected PERs associated with radwaste processing and transportation were reviewed. Three PERs and one Self-Assessment were reviewed in detail and discussed with HP supervision. 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. 4.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Cornerstone: Mitigating Systems

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting PIs, including Procedure SPP-3.4, Performance Indicator for NRC Reactor Oversight Process, for Compiling and Reporting PI's to the NRC. The inspectors reviewed raw PI data for the PI's listed below for the first quarter 2002 through the fourth quarter 2002. The inspectors compared graphical representations, from the most recent PI report to the raw data to verify that the data was correctly reflected in the report. The inspectors reviewed licensee procedure SPP 6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting - 10 CFR 50.65; category A and B PERs; engineering evaluations and associated PERs; and licensee records to verify that the PI data was appropriately captured for inclusion into the PI report, and the PI was calculated correctly. The inspectors reviewed Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revisions 1 and 2, to verify that industry reporting guidelines were applied.

- Unit 2 and Unit 3 Safety System Unavailability: RHR
- Unit 2 and Unit 3 Safety System Unavailability: Emergency AC Power Systems

b. Findings

No findings of significance were identified.

Cornerstone: Occupational Radiation Protection

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The licensee's records and data generated during Calendar Year (CY) 2002 for the Occupational Exposure Control Effectiveness PI were reviewed. The information reviewed included data reported to the NRC, pertinent corrective action program issues and selected Radiological Control program records. The inspectors assessed the licensee's CY 2002 monthly reviews for PI occurrences which were performed pursuant to procedure SPP-3.4, Performance Indicator for NRC Reactor Oversite Process, Rev. 0. The licensee's disposition of the reviewed issues was evaluated against NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 2. Specific procedures, records, and PERs reviewed and evaluated for this PI are identified in Section 4OA1 of the Attachment to this report.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Protection

.3 Radiological Effluent Control

a. Inspection Scope

The licensee's records and data generated during CY 2002 for the Radiological Effluent Control PI were reviewed. The information reviewed included data reported to the NRC, pertinent corrective action program issues and selected Radiological Effluent Control program records. The inspectors assessed the licensee's CY 2002 monthly reviews for PI occurrences which were performed pursuant to procedure SPP-3.4, Performance Indicator for NRC Reactor Oversite Process, Rev. 0. The licensee's disposition of the reviewed issues was evaluated against NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 2. Specific procedures, records, and PERs reviewed and evaluated for this PI are identified in the Attachment to the report.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Annual Sample Review

a. Inspection Scope

The inspectors selected two PERs, a maintenance/testing PER, 02-008512-000 (Level B), and maintenance/modification, 02-014617-000 (Level B), for detailed review. The maintenance/testing PER was associated with a HPCI valve failure during testing and the maintenance/modification was associated with a torus snubber being inoperable due to a maintenance error. The above and subsequent PERs were reviewed to verify the full extent of the issues were identified, an appropriate evaluation performed, and appropriate corrective actions were specified and prioritized. The PERs were evaluated against the requirements of the licensee's corrective action program (CAP) as delineated in the Standard Programs and Processes procedure SPP-3.1, Corrective Action Program, and the 10 CFR 50, Appendix B.

b. Findings and Observations

There were no findings identified associated with these reviewed samples involving corrective action, however, a licensee-identified finding involving maintenance procedures was noted. For the review of the maintenance/modification PER, 02-014617-000, the inspectors observed the licensee had replaced five Bergen-Patterson type snubbers with Lisega type snubbers. During the snubber changeout activity, the licensee initiated the following PERs for identified deficiencies:

- 99-004882-000 (Level C), identified the lack of an approved site procedure for maintenance of the Lisega type snubbers and also identified that a functional test specified in a work order was for Bergen-Patterson type snubbers and not Lisega type snubbers

- 99-005365-000 (Level C), identified the need for snubber inspection procedure enhancements, that a site maintenance procedure was needed, and also that a vendor manual instead of an approved site procedure was being used for maintenance on Lisega type snubbers

- 99-005399-000 (Level C), identified that there was no site surveillance procedure for the Lisega type snubbers, the adequacy of the Lisega type snubbers was based on testing performed by the vendor, and that a site surveillance procedure would be needed

- 00-007114-000 (Level C), identified that the site surveillance procedure did not specify acceptance criteria for Lisega snubber oil level or type of oil, and the Lisega snubber oil was not compatible with other types of oil

The licensee identified the following possible causes of the above PERs: a difference in interpretation of like-for-like material equivalency between engineering and maintenance that led to an inadequate impact review prior to the changeout of the snubbers;

engineering personnel thought that following the vendor manual instruction was acceptable and did not consider configuration control an issue; and the expectations for configuration control during mechanical maintenance activities were not clear. The inspectors verified the cause evaluations and short term associated corrective actions were appropriate and timely. The long term corrective actions were scheduled and have assigned completion dates. The licensee identified finding is discussed in Section 4OA7.

.2 Cross-References of PI&R Findings

Section 4OA7, Licensee Identified Violations, of this report describes a finding for failure to establish and implement a maintenance procedure for the Lisega type dynamic torus snubbers. The snubber procedure in use did not apply to the new Lisega snubber. The licensee failed to recognize this deficiency when five Lisega type snubbers, three in Unit 2 and two in Unit 3, were installed in April 1999. This failure resulted in a Unit 2 snubber being inoperable from the time of installation until October 2002.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 50-260/2002-003-00, Non-Conservative Oscillation Power Range Monitoring T-min Specification For Unit 2 and Unit 3

This LER reported a non-conservative condition affecting the Unit 2 and Unit 3 Oscillation Power Range Monitor (OPRM) system identified in a November 22, 2002, 10 CFR 21 report. The non-conservative condition was for the T-min value used in OPRM algorithm. The licensee determined the root cause was a flawed original vendor design of the OPRM system algorithm. The OPRM upscale trip function provides protection against exceeding the fuel minimum critical power ratio Safety Limit should thermal-hydraulic power oscillations occur. The licensee declared the OPRM upscale trip function on both units inoperable and completed the Technical Specification Limiting Condition of Operation to initiate an alternate method to detect and suppress thermal-hydraulic instability oscillations. The LER was reviewed by the inspectors and no findings of significance were identified. This event was entered in the licensee's corrective action program as PER 02-015282-000.

.2 Unit 2 Reactor Scram Due to Low Reactor Vessel Water Level

a. Inspection Scope

The inspectors responded to the control room and reviewed control room personnel's response to an unanticipated Unit 2 reactor scram, while the unit was shutdown, on February 24, due to a low reactor water level. The inspectors assessed operator performance to verify the requirements of licensee's procedures were met. The inspectors reviewed procedure 2-OI-3, Reactor Feedwater System, being used by operations to shut down the main feedwater system and align the condensate system for reactor water level control when the scram occurred, to assess procedure content and clarity. The inspectors also assessed the reactor feedwater system startup level control valve, 2-LCV-3-53, performance following apparent slow response when operators attempted to open the valve to realign feed flow. Procedures and records reviewed are documented in the Attachment to the report.

b. Findings

No findings of significance were identified

.3 (Closed) LER 259/85-33: Non-Standard Four-Inch Pipe Penetrations Through Secondary Containment

This LER reported temporary, non-standard, secondary containment penetrations in the Residual Heat Removal Service Water system tunnel of each unit. Engineering analysis determined that the penetrations were seismically qualified; however further work was required to fully qualify the penetrations for permanent use on Units 2 and 3. No further work was required for Unit 1. The licensee's design changes for Units 2 and 3 were previously reviewed and documented in NRC Inspection Reports, 50-259, 260, 296/88-28 and 50-259, 260, 296/95-26, closing the :LER for Units 2 and 3.

.4 (Closed) LER 259/94-03: Defect Discovered in Pressure Rating for Four Shutdown Board Room Air Conditioning Units Manufactured by Ellis and Watts

This LER reported a design error associated with the four air conditioning units (ACUs) installed for cooling the Unit 1 and Unit 3 Shutdown Board Rooms. The condensers were designed for 150 psig. However, under post accident conditions the condensers could experience an operating pressure up to 300 psig. The inspectors determined that the condensers for these ACUs are no longer used on either unit. All work to resolve this issue was accomplished under Design Change Notice W40283B as part of Unit 3 recovery. No further work was required for Unit 1.

.5 (Closed) Violation No. 50-259/95-03-01: Spring Can Installation

This item dealt with an incorrect spring can installed on Support No. 3-47B452-3035. The licensee's corrective actions for Unit 3 were reviewed and documented in NRC Inspection Report, 50-259, 260, 296/95-44. No work was required for Unit 1.

.6 (Closed): Inspector Followup Item (IFI) 50-259/95-19-02 Verify Valve Stator Through Bolts have been inspected or repaired.

The item related to the inspection and repair of valve motor stator through bolts for ten RHR system valves. The low strength bolts on the Unit 2 valves had become loose and failed due to RHR system vibrations. The licensee decided to replace all ten valve operators with new operators. The inspectors verified that WOs 02-016164-012, 02-016164-013, 02-016164-014, 02-016164-015, 02-016164-016, 02-016164-021, 02-016164-022, 02-016164-023, 02-016164-024, and 02-016164-025 had been issued to install the replacement valve operators during Unit 1 recovery. Therefore, since this item is effectively being tracked in the licensee's corrective action program, is being corrected identically to the Unit 2 solution with the same process and design change, and because any performance deficiencies associated with the licensee's corrective actions would have only minor consequences, this item meets closure crieria established for Unit 1 recovery issues. It is closed for Unit 1.

.7 Unit 2 Manual Scram Following Loss Of Forced Circulation

a. Inspection Scope

On March 26, the inspectors observed operator activities, monitored licensee controls over recovery activities, and reviewed selected items following a manual scram to verify that procedure and regulatory requirements were met. The scram was initiated following a trip of a second recirculating pump that resulted in a loss of forced flow reactor coolant. The first pump had tripped about two hours prior to the second pump trip. The inspectors also reviewed the licensee's post scram analysis and apparent cause of the pump trips and discussed immediate corrective actions with licensee management and engineering personnel to assess licensee performance. The licensee had implemented a major permanent modification during the outage to replace the reactor recirculation pump motor generator sets with new variable frequency drive units for both recirculation pumps. The licensee, with vendor assistance, determined that the vendor had failed to convert a trip function for a voltage circuit check to an alarm function that was specified in the design for the modification. The inspectors reviewed and discussed with licensee management the immediate corrective action to defeat the trip function and initiate an alarm function for the voltage circuit check. The inspectors compared their reviews and observations to 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-GOI-100-1A, Unit Startup and Power Operation to verify procedure requirements were met. Review and inspection of personnel performance is included in section 1R14 of the report.

b. Findings

No findings of significance were identified.

40A5 Other

(Closed) Unresolved Item (URI) 50-296/02-04-01, Troubleshooting Results in Unit 3 HPCI Failure.

Introduction: A Green self-revealing NCV of Technical Specification 5.4.1a was identified for an inadequate work control authorization process procedure and poor trouble shooting techniques that resulted in the failure of Unit 3 High Pressure Coolant Injection (HPCI) System.

Description: The licensee was performing troubleshooting on a battery charger supplying Battery Board #3 when a loss of the in-service charger occurred. Fuses were blown in the Unit 2 Division I and the Unit 3 Division II inverters. Although the loss of the Division I inverter resulted in a loss of instrumentation, control, and control room indication for specific systems on Unit 2, the impact was limited because the unit was in Mode 5 - Refueling at the time of the event. The impact on Unit 3 was the loss of the

Division II inverter resulting in a loss of torus temperature monitoring instrumentation, HPCI control and instrumentation, and turbine high water level trip logic, channel B. The total time, from power loss to restoration, for this condition on Unit 3 was 2 hrs, 25 minutes. The inspectors reviewed the SDP as it applied to the Unit 3 loss of HPCI for a time duration of 2 hrs, 25 minutes. The work control authorization procedure did not provide specific procedural direction for assessing all possible impacts of the proposed work activity on the associated plant systems and the troubleshooting techniques used to diagnose the problem exacerbated the problem.

Analysis; The finding is considered to be more than minor because the loss of HPCI had an actual impact on safety. Although the issue did present an immediate safety concern, the concern was promptly and appropriately addressed by immediate operator response actions. The finding was determined to affect the Mitigating System Cornerstone because it affected the operability, availability, reliability, or function of a system or train in a mitigating system (HPCI). The SDP Phase 1 screening worksheet was completed for the Mitigating Systems Cornerstone which concluded that a Phase 2 analysis was appropriate because the finding represented an actual loss of safety function of the HPCI system. During the current inspection period the NRC determined that the issue was of very low safety significance (Green). Some of the factors causing the issue to be of very low safety significance were:

- The plant power conversion systems, which were not credited in the SDP worksheets were all available
- The Reactor Core Isolation Cooling system, which was not credited in the SDP worksheets was available for core cooling
- All of the low pressure core cooling systems were available for injection

Enforcement: Because the failure to comply with TS 5.4.1a, inadequate work control authorization process procedure and poor trouble shooting techniques is of very low safety significance and has been entered into the licensee's corrective action program as PER 02-012529-000, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-296/03-02-01, inadequate work control process procedure and poor troubleshooting techniques resulted in the loss of the Unit 3 HPCI.

4OA6 Management Meetings

Exit Meeting Summary

.1 On 4/23/2003, the resident inspectors presented the inspection results to Mr. Ashok Bhatnager and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On April 3, 2003, the NRC's Chief of Reactor Project's Branch 6 and the Senior Resident Inspector assigned to the Browns Ferry Nuclear Power (BFNP) plant met with the Tennessee Valley Authority (TVA) to discuss the NRC's Reactor Oversight Process (ROP) and the Browns Ferry annual assessment of safety performance for the period of January 1, 2002 - December 31, 2002. The major topics addressed were: the NRC's assessment program, the results of the Browns Ferry assessment, and NRC security activities. Attendees included Browns Ferry site management, members of site staff, corporate management and staff, local public officials, and members of the local news media.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC' s document system (ADAMS) as accession number ML031130019. 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 Violation

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

- TS 5.4.1a requires, in part, that written procedures shall be established and implemented covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Contrary to this, on October 28, 2002, during a visual inspection of the Unit 2 Lisega torus dynamic snubber, 2-SNUB-064-0002, it was discovered that the internal 2-way ball valve was closed, due to not establishing and implementing a maintenance procedure for this type of snubber. The 2-way valve had apparently been closed since the snubber was installed in April, 1999. In order for the snubber to function as designed the internal ball valve must be in the open position. This was identified in the licensee's CAP as PER 02-014617-000. This finding, which impacts the Barrier Integrity Cornerstone, is of very low safety significance because it only affected one of four snubbers, did not result in making any system inoperable, and all mitigating systems were functional.

SUPPLEMENTARY INFORMATION
PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Abney, Nuclear Site Licensing & Industry Affairs Manager
A. Bhatnagar, Site Vice President
L. Clardy, Site Nuclear Assurance Manager
J. Corey, Radiation Protection and Chemistry Manager
R. Jones, Nuclear Plant Manager
J. Lewis, Nuclear Plant Operations Manager
T. Niessen, Jr., Engineering & Site Support Manager
J. Ogle, Site Security
R. Rogers, Maintenance & Modifications Manager
M. Skaggs, Assistant Plant Manager
R. Wiggall, Site Engineering Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened / Closed

50-296/03-02-01	NCV	Inadequate work control process procedure and poor troubleshooting techniques resulted in the loss of the Unit 3 HPCI (Section 40A5)

Closed

50-260/2002-03-00	LER	Non-Conservative Oscillation power Range Monitoring T-min Specification For Unit 2 and Unit 3 (Section 40A3)
50-296/02-04-01	URI	Troubleshooting Results in Unit 3 HPCI Failure (Section 40A5)
50-259/85-33	LER	Non-Standard Four-Inch Pipe Penetrations Through Secondary Containment (Section 40A3)
50-259/94-03	LER	Defect Discovered in Pressure Rating for Four Shutdown Board Room Air Conditioning Units Manufactured by Ellis and Watts (Section 40A3)
50-259/95-03-01	VIO	Spring Can Installation (Section 40A3)
50-259/95-19-02	IFI	Verify Valve Stator Through Bolts have been inspected or repaired (Section 40A3)

PARTIAL LIST OF DOCUMENTS REVIEWED

Section 1R05
Fire Hazards Analysis, Volume 1 and 2
Fire Protection Systems Limiting Conditions for Operating - 9.3.11.G, Fire-Rated Assemblies
Fire Protection Impairment Permit 03-00009 (Re: Problem Evaluation Report 03-000857-000
Fire Pre-Plans: RX2-621, RX3-621
Smoke Detector Locations: Procedure 0-SI-4.11.A.1(3)b
Drawings: 0-47W216-58, 0-47W216-57
Fire Hazards Analysis: Fire Areas 9, 14

Section 1R06
Procedures:

Procedure 0-AOI-100-3, Revision 25, Flood Above Elevation 558'
Mechanical Preventive Instruction (MPI)-0-260-DRS001, Revision 28, Inspection and Maintenance of Doors
MPI-0-000-INS001, Revision 7, Inspection of Flood Protection Devices
Electrical Preventive Instruction-0-077-SWZ002, Revision 3, Functional Check of the Reactor Building Flood Level Switches
Modification and Addition Instruction-3.4B, Revision 5A, Installation of Flood and Moisture Intrusion Seals

PERs/WOs:

2002-002394-000	2002-003007-000	2002-004897-000	2002-013104-000
2002-013305-000	2002-016666-000	2003-005186-000	2003-005170-000
2002-001612-000	2002-002019-000	2002-002019-001	2002-002019-002
2002-002019-003	2002-002394-000		

Section 1R08
NDE Procedure, N-MT-6, Magnetic Particle Examination for ASME and ASNI Code Components and Welds, Rev. 25
NDE Procedure, N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Rev. 24
NDE Procedure, N-UT-26, Ultrasonic Examination for the Detection of ID Pitting, Erosion, and Corrosion, Rev 21
NDE Procedure, N-UT-78, PDI Generic Procedure for the Manual Ultrasonic Examination of Reactor Vessel Welds, Rev. 2
NDE Procedure, N-RT-1, Radiographic Examination of Nuclear Power Plant Components, Rev. 25
NDE Procedure, N-VT-1, Visual Examination Procedure for ASME Section XI Preservice and Inservice, Rev 33
Surveillance Instruction, 2-SI-4.6.G, Inservice Inspection and Risk-Informed Inservice Inspection Program, Unit 2, Rev 20
Technical Instruction, 0-TI-140, Monitoring Program for Flow Accelerated Corrosion, Rev 0
Technical Instruction, 0-TI-363, ASME Section XI Repair and Replacement, Rev. 5
Technical Instruction, 0-TI-365, Reactor Pressure Vessel Internals Inspection, Units 1, 2 and 3, Rev. 10
TVA Standard, SPP-9.1, ASME Section XI, Rev. 4
TVA Standard, SPP-9.7, Corrosion Control Program, Rev 4
Problem Evaluation Report (PER) 99-001030-000, Documents recommendations contained in

EPRI assessment of TVA NDE and ISI programs as applicable to BFN
PER 03-004060-000, Jet Pump 17 abnormal; wear on wedge bearing surface EPRI Assessment of TVA NDE and ISI Program, 1998
GE Report DRF 000-0014-1252, Qualitative Assessment of Jet Pump Wedge Damage at BFNP Unit 2, March 8, 2003
Work Order (WO) 02-011110-000, Ultrasonic examination of Unit 1 main steam and feedwater piping in drywell
WO 02-011160-000, Ultrasonic examination of Unit 1 main steam piping in turbine building

Unit 2 Reports

UT Report	Component
R027	ASME Class 2 RHR piping weld, RHRG-2-07-A
R058	ASME Class 2 HPCI piping weld, THPCI-2-109
R059	ASME Class 2 HPCI piping weld, THPCI-2-107
R110	ASME Class 1 RPV Head Vent Nozzle N6A weld
R141	ASME Class 1 Feedwater to RPV Nozzle N4A weld
R142	ASME Class 1 Feedwater to RPV Nozzle N4B weld
R143	ASME Class 1 Feedwater to RPV Nozzle N4C weld
R144	ASME Class 1 Feedwater to RPV Nozzle N4D weld
R145	ASME Class 1 Feedwater to RPV Nozzle N4E weld
R146	ASME Class 1 Feedwater to RPV Nozzle N4F weld

MT Report	Component
R021	ASME Class 2 RBCCW piping weld attachment, 2-47B4650228-IA
R031	ASME Class 2 Core Spray piping weld attachment, CS-2-H-7-IA
R036	ASME Class 2 Core Spray piping weld attachment, CS-2-H-11-IA

PT Report	Component
R052	ASME Class 1 Recirc System piping weld RWR-2-003-146
R053	ASME Class 1 Recirc System piping weld RWR-2-001-184
R071	ASME Class 1 Recirc System piping weld RWR-2-003-145
R072	ASME Class 1 Recirc System piping weld RWR-2-003-183
R148	ASME Class 1 RPV Stabablizer weld attachment, RPV-STAB-2-1A-IA
R149	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1B-IA
R150	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1C-IA

R151	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1D-IA
R152	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1E-IA
R153	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1F-IA
R154	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1G-IA
R155	ASME Class 1 RPV Stabilizer weld attachment, RPV-STAB-2-1H-IA

Weld Examination Reports

Weld RADW-2-001-002	Three inch ASME Class II Drywell floor drain piping
Weld CRD-2-009-060	Two inch ASME Class II CRD Scram Discharge piping
Weld CRD-2-009-061	Two inch ASME Class II CRD Scram Discharge piping weld
Weld CRD-2-009-079	Two inch ASME Class II CRD Scram Discharge piping weld
Weld CRD-2-009-080	Two inch ASME Class II CRD Scram Discharge piping weld

Unit 1 UT Examination Records

Piping Segments	Component
1-MS-A-1 through 7	26 inch MS Line B Piping in Drywell
1-MS-B-8 through 14	26 inch MS Line C Piping in Drywell
1-MS-C-15 through 20	26 inch MS Line A Piping in Drywell
1-MS-D-21 through 26	26 inch MS Line D Piping in Drywell
1-MS-A-27 through 28	24 inch MS Line A Piping in Turbine Building
1-MS-B-29 through 30	24 inch MS Line B Piping in Turbine Building
1-MS-C-31 through 32	24 inch MS Line C Piping in Turbine Building
1-MS-D-33 through 34	24 inch MS Line D Piping in Turbine Building
1-FW-A-1 through 5	24 inch FW Piping in Drywell
1-FW-B-13 through 17	24 inch FW Piping in Drywell
1-FW-A-6	20 inch FW Piping in Drywell
1-FW-B-11	20 inch FW Piping in Drywell
1-FW-A-5 through 8	12 inch FW Piping in Drywell
1-FW-A-9 through 12	12 inch FW Piping in Drywell

Section 1R20

UFSAR Section 10.22, Auxiliary Decay Heat Removal System
0-AOI-72-1, Auxiliary Heat Removal System Failures
Equipment clearance, 2-TO-2003-0001-02793, EECW
Equipment clearance, 2-TO-2003-0001-01513, EDG A

<u>Section 2OS1</u>

Procedures, Instructions, Lesson Plans, and Manuals

Standard Programs and Processes (SPP)-5.1, Radiological Controls, (Rev.) 4
Radiological Control Instruction (RCI)-1.1, Field Operations Program Implementation, Rev. 103
RCI-2.1, External Dosimetry Program Implementation, Rev. 51
RCI-8.1, Internal Dosimetry Program Implementation, Rev. 33
RCI-9.1, Radiation Work Permit Preparation and Administration, Rev. 34
RCI-17, Control of High Radiation Areas and Very High Radiation Areas, Rev. 42
Radiological Control Department Procedure (RCDP)-6, Special Dosimetry Operations, Rev. 2
RCDP-7, Bioassay and Internal Dose Program, Rev. 0

Radiation Work Permits (RWPs)

0322002 Maintenance on Miscellaneous Systems
03270782 RX-2 Cycle 12 Maintenance on Fuel Pool Cooling System
03270784 RX-2 Cycle 12 Maintenance on Fuel Pool Cooling System (Dose Control)
03273192 Unit 2 Cycle 12 RHR Keep Fill
03280745 DW-2 Cycle 12 Maintenance on RHR System (Full Face Respirator)
03280746 DW-2 Cycle 12 Maintenance on RHR System (Supplied Air Respirator)
03290005 RX-2 Cycle 12 Vessel Disassembly, Fuel Shuffle, Reassembly

Records and Data

Radiation Survey 020503-1, Unit 3 RXB 621' General Area (2/5/2003)
Radiation Survey 021903-6, Unit 2 RXB 639' General Area (2/19/2003)
Radiation Survey 022103-7, Unit 2 RXB 621' Fuel Pool Cooling Area (2/21/2003)
Radiation Survey 021903-4, Unit 2 RXB 621' General Area (2/19/2003)
Radiation Survey 03-10305-07, RX2, 621' FPC Work Area (2/27/2003)
Internal Dose Assessment 031200490 (3/2/2003)
Internal Dose Assessment 031200548 (3/4/2003)

Audits, Self-Assessments, and Problem Evaluation Reports (PERs)

Self Assessment BFN-RP-02-005, Radiation Exposure Control (9/9-27/2002)
Self Assessment BFN-RP-01-002, Radiation Exposure Control (1/29-2/2/2001)
PER 03-001661-000, Work Crew Was Briefed for Work in Radiation but Entered a High Radiation Area. Crew Exited Area and Reported to Radcon after Receiving Dose Rate Alarm, (1/30/2003)
PER 03-000696-000, Local Audible Alarms for Some Locked High Radiation Areas Are Difficult to Hear in High Noise Environments, (1/15/2003)
PER 03-000579-000, High Radiation Area Rope Boundary Found Unattached (1/13/2003)
PER 03-000100-000, Contractor Employee Failed to Exchange TLD for New Quarter (1/5/2003)

PER 02-012777-000, Worker Briefed on Improper RWP. Worker Received Dose Rate Alarm, Exited Area and Reported to Radcon (10/27/2002)

Section 2OS2

Procedures, Instructions, Lesson Plans, and Manuals

General Operation Instruction, 2-GOI-100-12A, Unit Shutdown From Power Operation to Cold Shutdown and Reductions in Power During Power Operations, Rev. 64

RCI-15.1, Maintaining Occupational Radiation Exposures As Low as Reasonably Achievable (ALARA), Rev. 27

RCI-15.2, Temporary Shielding, Rev. 17

RCI-15.3, ALARA/Radwaste Committee, Rev. 11

RCI-15.4, ALARA/Radwaste Volume Reduction Suggestion Program, Rev. 4

RCI-2.1, External Dosimetry Program Implementation, Rev. 51

RCI-9.1, Radiation Work Permit Preparation and Administration, Rev. 34

SPP-5.2, ALARA Program, Rev. 1

ALARA Packages

Revised ALARA Packages - Current Outage:

ALARA Planning Report, 03-0034, Rev. 2, U2C12 Outage - Replace SDIV Level Switches and Support Work.

ALARA Planning Report, 03-0037, Rev 1, U2C12 Outage - Repair, Maintenance, Testing and SI's on Various Systems in Turbine Building

ALARA Planning Report, 03-0025, Rev. 1, U2C12 Snubber Maintenance

ALARA Planning Report, 03-0021, Rev. 1, U2C12 Outage - Scaffolding, Insulation and Shielding Installation, Maintenance, Removal

ALARA Planning Report, 03-0035, Rev. 2, U2C12 Outage - Outage Recirc Drains and Equipment Mods

ALARA Planning Report, 03-0023, Rev. 1, U2C12 Outage - RadCon/RadWaste Support

Final ALARA Packages - Prior Outage:

ALARA Planning Report, 02-0019, Rev. 1, U3C10 Outage - Vessel Disassembly/Fuel Shuffle/Invessel work

ALARA Planning Report, 02-0021, Rev. 0, U3C10 Outage - ISI, IWE, FAC, and Engineering Support

ALARA Planning Report, 02-0022, Rev. 2, U3C10 Outage - Scaffolding, Insulation and Shielding Installation, Maintenance, Removal

ALARA Planning Report,02-0023, Rev. 1, U3C10 Outage - Mechanical, Electrical, and Instrument Maintenance - Inspections, low dose maintenance, walkdowns and surveillances

ALARA Planning Report, 02-0028, Rev. 1, U3C10 Outage - RadCon/RadWaste Support

Records

U2C12 Outage Report March 11, 2003, Day 16

Audits and Self-Assessments and PERs

Radiation Exposure Control Self Assessment Report, BFN-RP-01-002,1/29/2001-2/2/2001

Radiation Exposure Control Self Assessment Report, BFN-RP-02-005, 9/9-27/2002

Fiscal Year 2001and 2002 Annual ALARA Report

PER 02-001199-000, 2/6/02, Actual Dose Received Was Greater than Estimated Due to Higher than Expected Dose-Rates. 21 vs 16 Mrem

PER 02-003434-000, 3/31/02, Two Mechanical Maintenance Employees Were Contaminated

During the Removal of 3-RF-069-0533 (Tube Side Relief Valve) for Testing on WO-00-007747-005.

PER 02-000759-000, 1/24/02, Actual Dose Received Was Much less than Expected. Dose Rate and Man Hours Were Both Much less than Expected.

PER 02-000758-000, 1/24/02, Actual Dose Received Was Much less than Expected. Dose Rate and Man Hours Were Both Much less than Expected.

PER 02-016574-000, 12/17/02, Dose Accrued on the Unit 1 Restart Project Exceeded the Dose Goals for the Month of November (2002). The Dose Goal for November Was Set at 10.286 Man-rem. The Dose Goals for the Month of November Was Set at 10.286 Man-rem. The Total Dose Accrued Was 11.465 (Work Schedule Had Been Accelerated.)

PER 02-011864-000, 10/7/02, During Divisional Outages on the RHR System, Craft Move Scaffolding and Equipment from the South West Quad to Southeast Quad, and Back Again.

PER 02-001633-000,2/16/02, 3a and 3b RWCU Pump Oil Usage/Leakage Is Excessive. This Is a on Going Issue That Is an ALARA Concern to the Individuals That Have to Add Oil and Radcon That Has to Assist.

<u>Section 2PS2</u>

Procedures, Manuals, and Guides

SPP-3.1, Corrective Action Program, Rev. 4.

0-Operating Instruction (OI)-77E, Solid Radwaste, Rev. 31.

Radioactive Material Shipment Manual, Rev. 35, Volumes I, II, and III.

Radioactive Waste Technical Procedure (RWTP)-100, Radioactive Material/Waste Shipments, Rev. 0.

RWTP-101, 10 CFR 61 Waste Characterization, Rev. 0.

RWTP-102, Use of Casks, Rev. 0.

RCI-1.1, Field Operations Program Implementation, Implementing Procedure (IP)-4, Rev. 103, and IP-9, Rev. 102.

Duratek Procedure TR-OP-030, Handling Procedure for Transport Cask Number CNS 14-170 Series III, Rev. 10.

Environmental/Waste Control - Section Instruction Letter EWC-1, Radwaste Controller Qualification Program, 01/12/2000.

Shipping Records and Radwaste Data

020110, Low Specific Activity (LSA) II, Plant Wet Sludge, 01/30/02.

020903, LSA, II, Contaminated Laundry, 09/04/02.

020913, Surface Contaminated Object (SCO) II, Uncoupling Tool, 09/26/02.

030107, LSA II, DAW, 01/09/03.

020510, Type A, Coupon Samples, 05/29/02.

10 CFR 61 Analysis Report, 1999, 2000, 2001, 2002.

PERs and Audits

PER 02-001384-000, Changes Made to Liquid Radwaste System via Work Order, 02/08/02.
PER 02-009878-000, Contaminated Motor Transported on Public Road, 08/23/2002.
PER 03-003027-000, Contamination Found on Clean Shipment from Vendor Lab, 02/23/03.
Self-assessment CRP-ENV-02-003, Radioactive Waste Program Performance, 07/08/02 - 08/09/02.

Requirements Documents

Browns Ferry Nuclear Plant, Updated Final Safety Analysis Report, Amendment 19, Chapter 9, Radioactive Waste Control Systems.
Process Control Program Manual (PCP), Rev. 2.

Section 4OA1

Procedures

SPP-3.4, Performance Indicator for NRC Reactor Oversight Process, Rev. 0
Desktop Guide for Identification and Reporting of NEI 99-02, Rev. 2 Performance Indicators for Occupational Exposure Control Effectiveness
CI-138, Reporting NEI Indicators, Rev. 1
0-SI-4.8.8b.3, Appendix I Dose Calculations-Airborne Effluents, Rev. 19

Records

CY 2002 Monthly Memos from the Radiological Protection Supervisor Reporting Occupational Exposure Occurrences.
CY 2002 Monthly Access Control Alarm Reports
CY 2002 Monthly Radioactive Effluent Release Reports

Section 40A3.2

2-OI-3, Reactor Feedwater System
2-EOI-1, Emergency Procedure for Reactor Pressure Vessel Control
General Operating Instruction, 2-GOI-100-12A, Unit Shutdown From Power Operation to Cold Shutdown and Reduction in Power During Power Operation
Abnormal Operating Instruction, 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low
Abnormal Operating Instruction, 2-AOI-100-1, Reactor Scram
General Operating Instruction, 2-GOI-100-1A, Unit Startup and Power Operation