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

January 29, 2007

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

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2006005 AND 05000391/2006005

Dear Mr. Singer:

On December 31, 2006, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant, Units 1 and 2. The enclosed integrated inspection report documents the inspection results which were discussed on January 10, 2006, with Mr. M. Skaggs and other members of your staff.

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

Based on the results of this inspection, no findings of significance were identified. However, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. NRC is treating this violation as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance of the violation and because it is entered into your corrective action program. If you contest this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington D.C. 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D. C. 20555-0001; and the NRC Resident Inspector at the Watts Bar Nuclear Plant.

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

Sincerely,

/RA/

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

Docket Nos. 50-390, 50-391
License No. NPF-90 and Construction
Permit No. CPPR-92

Enclosure: NRC Inspection Report 05000390/2006005, 05000391/2006005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to Karl W. Singer from Malcolm T. Widmann dated January 29, 2007

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000390/2006005 AND 05000391/2006005

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-390, 50-391
License Nos:	NPF-90 and Construction Permit CPPR-92
Report Nos:	05000390/2006005, 05000391/2006005
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	Spring City, TN 37381
Dates:	October 1, 2006 - December 31, 2006
Inspectors:	R. Baldwin, Senior Operations Engineer (Section 1R11.1) J. Bartley, Senior Resident Inspector M. Bates, Senior Operations Engineer (Section 1R11.1) W. Fowler, Reactor Inspector (Section 4OA5.1) J. Fuller, Reactor Inspector (Sections 1R08, 4OA5.3) J. Kreh, Emergency Preparedness Inspector (Section 2OS1) L. Lake, Reactor Inspector (Sections 1R08) W. Loo, Senior Health Physicist (Sections 2OS2, 2PS2) B. Miller, Reactor Inspector (Sections 1R08) M. Pribish, Resident Inspector S. Vias, Senior Reactor Inspector (Sections 1R08, 4OA5.3) R. Taylor, Reactor Inspector (Section 4OA5.1)
Approved by:	Malcolm T. Widmann, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000390/2006005, 05000391/2006005; 10/01/2006 - 12/31/2006; Watts Bar, Units 1 & 2; Routine Integrated Report.

The report covered a three-month period of routine inspection by resident inspectors and announced inspections by a regional senior health physicist, a regional emergency preparedness inspector, two regional senior operations engineers and six regional reactor inspectors. No findings of significance were identified. 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. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 was in the Unit 1 Cycle 7 (U1C7) refueling and steam generator(SG) replacement outage at the start of the inspection period. The unit completed the outage on November 30, 2006, and reached full power on December 13, 2006. The unit operated at or near 100 percent power for the remainder of the inspection period. Unit 2 remained in a suspended construction status.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Extreme Weather Readiness

a. Inspection Scope

The inspectors reviewed licensee actions taken in preparation for low temperature weather conditions to limit the risk of freeze-related initiating events and to adequately protect mitigating systems from its effects. The inspectors reviewed the licensee procedure 1-PI-OPS-1-FP, Freeze Protection, and walked down selected components associated with the four areas listed below to evaluate implementation of plant freeze protection, including the material condition of insulation, heat trace elements, and temporary heated enclosures. In addition, the inspectors reviewed the licensee's response during an actual freezing condition on December 4, 2006. Corrective actions for items identified in relevant problem evaluation reports (PERs) and work orders (WOs) were assessed for effectiveness and timeliness. Documents reviewed are listed in the attachment to this report.

- Refueling Water Storage Tank (RWST) freeze protection preparations
- A and B-train essential raw cooling water (ERCW) system freeze protection preparations
- A and B-train high pressure fire protection system freeze protection preparations
- Main feedwater sensing lines freeze protection preparations

b. Findings

No findings of significance were identified.

.2 Readiness for Impending Adverse Weather Condition

a. Inspection Scope

The inspectors reviewed the licensee's preparation for and response to an actual freezing condition on December 4, 2006. The inspectors verified performance and reviewed the data associated with temperature monitoring of the RWST, which is required per licensee procedure 1-PI-OPS-1-FP for outside air temperature less than 25°F. In addition, the inspectors performed a walkdown of the RWST freeze protection enclosures to verify the adequacy of construction and the operation of the installed temporary lighting.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

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

- B-train diesel generators (DGs) while A-train DGs removed from service for A-train essential raw cooling water (ERCW) header outage
- B-train auxiliary air system during A-train auxiliary air system maintenance
- B-train high pressure fire protection (HPFP) during A-train HPFP maintenance

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of 10 areas important to reactor safety, listed below, to verify the licensee's implementation of fire protection requirements as described in the Fire Protection Program, Standard Programs and Processes (SPP)-10.0, Control of Fire Protection Impairments; SPP-10.10, Control of Transient Combustibles; and, SPP-10.11, Control of Ignition Sources (Hot Work). The inspectors evaluated, as appropriate, conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment, and features; and, (3) the fire barriers used to prevent fire damage or fire propagation.

- Cable spreading room
- 480 V Reactor Motor Operated Valve (MOV) Board Rooms 1A, 1B, 2A, and 2B
- Vital Battery Rooms I, II, III, IV, and V

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Semiannual Internal Flood Protection Inspection

a. Inspection Scope

The inspectors reviewed internal flood protection measures for the DG building. Flooding in a single DG room could impact the other DGs if flood mitigation features became degraded. DG building flood protection features were examined to verify that they were installed and maintained consistent with the plant design basis. The inspectors conducted walkdowns of the DG building to verify the following attributes: (1) sealing of equipment below the flood line, such as electrical conduits; (2) holes or unsealed penetrations in floors and walls between flood areas; (3) common drain system and sumps, including floor drain piping and check valves; and (4) sources of potential internal flooding that are not analyzed or not adequately maintained, such as failure of high pressure fire protection piping. In addition, the inspectors reviewed the licensee's corrective action program (CAP) to ensure that the licensee was identifying flood-related problems and that they were properly addressed for resolution. Documents reviewed are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.2 Annual External Flood Protection Inspection

a. Inspection Scope

The inspectors reviewed licensee flood analysis documents to identify design features important to external flood protection and areas that can be affected by flooding; design flood levels; and protection features for areas containing safety-related equipment, such as level switches and sumps. The inspectors reviewed licensee procedures to cross-tie systems in the event of severe flooding to verify that procedures could be accomplished as written and that selected spool pieces and tools were staged per procedure. The inspectors walked down the lower level of the intake pumping structure to observe material condition of its flooding protection features such as doors, floor drains, sump level switches, and sump pumps. In addition, the inspectors reviewed the corrective action program documents to verify that the licensee was identifying and resolving flooding protection issues. Documents reviewed are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems and Containment ISI

a. Inspection Scope

From October 2 - November 2, 2006, the inspectors observed and reviewed the licensee's implementation of their ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and other risk-significant piping system boundaries for Watts Bar Unit 1. The inspectors observed and reviewed a sample of American Society of Mechanical Engineers (ASME), Section XI, Section III, and Risk-Informed ISI required examinations, in order of risk priority, as identified in Section 71111.08-03 of Inspection Procedure 71111.08, Inservice Inspection Activities.

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate compliance with TS and the applicable editions of ASME Section V and XI to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI acceptance standards.

The inspectors observed the following examinations:

Manual Ultrasonic Testing:
- Weld Number RHRF-D055-11, 8" diameter valve to pipe weld, ASME Class 1

Liquid Penetrant Examination:
- Weld Number 1-003B-B160-03, 3/4" Steam Generator #2 lower instrumentation tap “D,” ASME Class 2
- Weld Number 1-003B-B160-04, 3/4" Steam Generator #2 lower instrumentation tap “D,” ASME Class 2
- Weld Number 1-003B-B195-02, 3/4" Steam Generator #2 lower instrumentation tap B, ASME Class 2
- Weld Number 1-003B-B195-03, 3/4" Steam Generator #2 lower instrumentation tap B, ASME Class 2
- Weld Number 1-003B-B166-03, 3/4" Steam Generator #2 lower instrumentation tap “C,” ASME Class 2
- Weld Number 1-003B-B166-04, 3/4" Steam Generator #2 lower instrumentation tap “C,” ASME Class 2

- Weld Number 1-003B-B196-03, 3/4" Steam Generator #3 lower instrumentation tap B, ASME Class 2
- Weld Number 1-003B-B196-04, 3/4" Steam Generator #3 lower instrumentation tap B, ASME Class 2

The inspectors reviewed the following examination records:

Radiography Examination:
- Weld Number 1-10, 4"-RC-14-1502, Pressurizer Spray Line, ASME Class 2
- 1-SCV-B023-01, Steel Containment Vessel (SGRP),Plate to Plate, ASME III, NE
- 1-003B-B002-17A, Feedwater - Elbow to Pipe, ASME III Class 2
- 1-003B-B002-17B, Feedwater - Elbow to Pipe, ASME III Class 2
- 1-003B-B002-08B, Feedwater - Elbow to Elbow, ASME III Class 2

Liquid Penetrant (PT) Examination:
- FSK-—156, FW-50, Residual Heat Removal (RHR) slip on flange, ASME Section III Class 2
- FSK-—200, 31" Inside diameter RCS Cold Leg for Steam Generator (SG) #3, Butt weld prep, ASME Class 1
- FSK-—198, 31" Inside diameter RCS Hot Leg for SG#1, Butt weld prep, ASME Class 1
- FSK-—017, SG#4 Shell Drain Nozzle End Preps, ASME Class 2
- Weld Number 1-068F-B001-01, SG#1 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068F-B003-01, SG#3 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B004-02, SG#3 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B001-02, SG#1 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B002-02, SG#2 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068F-B002-01, SG#2 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B005-02, SG#4 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068F-B004-01, SG#4 RCS Piping, 31" Inside Diameter Nozzle to Elbow, ASME Section III, Class 1

Visual Examination (VT):
- VT-2: Remote Visual Examination of outside surface of Reactor Pressure Vessel (RPV) Closure Head and Penetration, 9/2006
- VT-2: Remote Visual Examination of outside surface RPV Lower Head bottom mounted instrumentation penetrations, 9/2006

Magnetic Particle Examination:
- FW-1 for Loop 2 Upper Lateral Support, ASME Section III

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

- VT-3: RHRHXH-2-1A, 1A RHR Heat Exchanger support
- PT: SG#4 Shell Drain Nozzle End Preps, ASME Class 2
- UT: Weld Number RHRF-D055-11, 8" diameter valve to pipe weld, ASME Class 2
- UT: Wall thickness measurement, 1-003B-B002-084, Feed Water, ASME Class 2
- MT: FW-1 for Loop 2 Upper Lateral Support, ASME Section III

The inspectors observed in-process welding activities for the following ASME pressure boundary locations. Inspectors reviewed quality records for welding procedures, procedure qualification, welder qualification, and filler metal certification.

The inspectors observed a sample of in-process welding activities for the following welds:

- Weld Number 1-068F-B003-01-c1, Reactor Coolant System cold leg pipe to SG3 nozzle, ASME Class 1
- Weld Number 1-068F-B002-01-c1, Reactor Coolant System cold leg pipe to SG2 nozzle, ASME Class 1
- Weld Number 1-068D-B001-02-c1, Reactor Coolant System hot leg pipe to SG1 nozzle, ASME Class 1
- Weld Number 1-068D-B002-02-c1, Reactor Coolant System hot leg pipe to SG4 nozzle, ASME Class 1
- Weld Number 1-068F-B001-01, Reactor Coolant System cold leg pipe to SG1 nozzle, ASME Class 1
- Weld Number 1-SCV-B023-1, 2/3 Side of Steel Containment Vessel, ASME Class 3
- Weld Number 1-SCV-B014-01, 1/4 Side of Steel Containment Vessel, ASME Class 3

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Upper Head Penetrations

a. Inspection Scope

The inspectors reviewed the licensee’s reactor pressure vessel (RPV) head and vessel head penetration (VHP) nozzle inspection activities that were implemented in accordance with the requirements of Order EA-03-009 (NRC Accession Number

ML040220391), issued on February 20, 2004. The licensee was not required to conduct non-visual examinations of the RPV head during this outage.

The inspectors reviewed tapes of the licensee's remote bare metal visual examination records for both 100% of the RPV upper head and head penetrations and the bottom head, bottom-mounted instrumentation penetrations. The inspectors evaluated licensee compliance with the requirements of NRC Order EA-03-009 for implementing procedures and applicable industry guidance. In particular, the inspectors verified that the licensee's visual examinations focused on locations where boric acid leaks could cause degradation. In addition, the inspectors reviewed recordable indications of penetrations, personnel and procedure qualifications, and training records.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Program (BACCP)

a. Inspection Scope

The inspectors reviewed the licensee's Boric Acid Corrosion Control Program to ensure compliance with commitments made in response to NRC Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary, and NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity.

The inspectors conducted an on-site record review and an independent walkdown of the auxiliary and reactor buildings. The inspectors evaluated licensee compliance with program procedures and applicable industry guidance. In particular, the inspectors verified that the licensee's visual examinations focused on locations where boric acid leaks could cause degradation of safety-significant components and, that degraded or non-conforming conditions were properly identified in the licensee's CAP. The inspectors reviewed documentation for the visual examination of reactor pressure vessel bottom-mounted instrumentation, inspection of insulated bolted connections and principal leak locations, and reactor coolant system pressure tests.

The inspectors reviewed a sample of engineering evaluations completed for boric acid on RCS piping and other ASME Code class components to verify that the minimum design code required section thickness had been maintained for affected components. The inspectors also reviewed licensee corrective action documents initiated for boric acid related issues to confirm that they were consistent with the requirements of Section XI of the ASME Code, 10CFR 50 Appendix B Criterion XVI, and licensee BACCP procedures. Specifically, the inspectors reviewed engineering evaluations for boric acid identified on the following components:

- 2-ISV-62-1053A, Boric Acid Transfer Pump Discharge Isolation Valve
- 1-CKV-62-658 S, Normal Charging Bypass Check Valve

- 1-STRU-661-5000, Fuel Transfer Canal Joints
- 1-ISIV-68-445B/3, Test Tee Fitting on Sense Line
- 1-TV-62-611, Pipe Cap on Reactor Coolant Pump #1 Seal Water Injection Test Line

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

Watts Batts SGs were replaced during this outage; therefore, no eddy current inspections were done during the regular outage inspection. The inspectors did conduct a review of the replacement SGs pre-service inspection activities in February 2006. This review is documented in Inspection Report 05000390, 05000391/2006002, paragraph 4OA5.1, issued April 14, 2006.

The inspectors reviewed the eddy current data, procedures, and programs to verify compliance with NRC Regulatory Guide 1.83, TVA Watts Bar TS, and Sections V and XI of the 1995 ASME Code through the 1996 Addenda and to determine that SG tube eddy current testing (ECT) examination scope was sufficient to identify tube degradation and to confirm that the ECT scope was completed. Additionally, the inspectors reviewed the SG tube ECT examination scope to determine that it was consistent with that recommended in Electrical Power Research Institute (EPRI) Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6.

b. Findings

No findings of significance were identified.

.5 Problem Identification and Resolution

a. Inspection Scope

The inspectors performed a review of ISI-related problems that included welding, boric acid corrosion control, and SG replacement activities that were identified by the licensee and entered into their CAP. The inspectors reviewed a sample of these corrective action documents to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Biennial Review

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of August 14, 2006, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of simulator operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR 55, Operators' Licenses. The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, Operator Licensing Examination Standards for Power Reactors, and Inspection Procedure 71111.11, Licensed Operator Requalification Program. The inspectors also reviewed and evaluated the licensee's simulation facility for adequacy in operator licensing examinations. The inspectors observed two operator crews during the performance of the operating tests. Documentation reviewed included written examinations, job performance measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records and performance test records, the feedback process, licensed operator qualification records, remediation plans, watchstanding, and medical records. The records were inspected against the criteria listed in Inspection Procedure 71111.11. Documents reviewed during the inspection are listed in the attachment to this report.

b. Findings

Introduction: The inspectors identified an unresolved item (URI) associated with multiple operators who potentially reactivated their reactor operator (RO) and senior reactor operator (SRO) licenses in accordance with Procedure OPDP-1, Conduct of Operations, incorrectly.

Description: During the biennial requalification inspection, the inspectors identified that RO/SRO licenses were reactivated without following the guidance in Procedure OPDP-1, which required a plant tour with a licensed operator that included those areas covered by shift rounds. OPDP-1 implemented the regulatory requirements of 10 CFR 55.53(f) which specifies that in order to activate an operator license, a complete tour of the plant shall be completed with a licensed operator.

The inspectors compared the information contained in four reactivation records with vital area computer access records to verify the plant tours were performed as indicated. The inspectors identified that two operators had potentially not satisfied the requirements for conducting a complete plant tour as part of reactivating their RO or SRO license respectively. The licensee was not able to provide documentation, in the form of security card reader records, that would confirm that an RO performed his tour in the presence of an active RO or SRO licensed individual. Also, records supplied by the licensee showed that an SRO who was reactivating his license performed his tour with an SRO whose license was not currently active.

Analysis: The potential failure to properly reactivate an RO or SRO license could impact the reactivating individual's knowledge of current plant status. The issue is associated with the mitigating systems cornerstone, because it deals with human performance attributes that affect the availability, reliability, and capability of licensed operators to respond to initiating events to prevent undesirable consequences.

Enforcement: 10 CFR 55.53(f), Conditions of a License, states, in part, that the licensee has completed a minimum of 40 hours of shift functions under the direction of an operator or senior operator as appropriate and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the plant and all required shift turnover procedures. To ensure 10 CFR 55.53 (f) requirements are satisfied, the licensee has established Procedure OPDP-1, Conduct of Operations Attachment P, which states, "The licensee has completed a minimum of 40 hours of shift functions under the directions of an active RO or SRO and in the position to which the operator is to be assigned. The 40 hours must include a complete plant tour (with an active licensed operator) of the plant and a review of all required shift turnover procedures. This tour shall be in those areas covered by shift rounds."

TS 5.7.1.1(a) requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Administrative Procedures. OPDP-1 is a written procedure that is required by TS 5.7.1.1(a).

On August 18, 2006, the inspectors identified that the licensee potentially failed to properly reactivate two licensed operators on May 20, 2005, and January 21, 2006, in accordance with procedure OPDP-1. The licensee entered this issue into their CAP as PER 108957. This item is identified as URI 05000390/2006005-01, Potential failure to properly reactivate RO/SRO licenses, and is unresolved pending additional NRC review of the circumstances and information regarding the reactivation of these two licensed operators.

.2 Resident Inspector Quarterly Review

a. Inspection Scope

On December 15, 2006, the inspectors observed operators in the plant's simulator during simulator scenario 3-OT-SRT-E1-4, Loss of Coolant Accident, Pressurizer Safety Fails Open, to verify operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with procedures TRN-1, Administering Training, and TRN-11.4, Continuing Training for Licensed Personnel. In addition, the inspectors verified that the training program included risk-significant operator actions, emergency plan implementation, and lessons learned from previous plant experiences.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two performance-based problems listed below. The focus of the reviews was to assess the effectiveness of maintenance efforts that apply to scoped structures, systems, or components (SSCs) and to verify that the licensee was following the requirements of TI-119, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting 10 CFR 50.65, and SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting 10 CFR 50.65. Reviews focused, as appropriate, on (1) appropriate work practices; (2) identification and resolution of common cause failures; (3) scoping in accordance with 10 CFR 50.65; (4) characterization of reliability issues; (5) charging unavailability time; (6) trending key parameters; (7) 10 CFR 50.65 (a)(1) or (a)(2) classification and reclassification; and (8) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). Documents reviewed are listed in the attachment to this report.

- B-train shutdown board room (SDBR) chiller condenser temperature control valve required manual adjustment for continued operation
- Containment sump to B-train containment spray pump suction flow control valve, 1-FCV-72-45, failed to open electrically during surveillance testing

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated, as appropriate for the two work activities listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and, (4) that maintenance risk assessments and emergent work problems were adequately identified and resolved. The inspectors verified that the licensee was complying with the requirements of 10 CFR 50.65 (a)(4); SPP-7.0, Work Control and Outage Management; SPP-7.1, Work Control Process; and TI-124, Equipment to Plant Risk Matrix.

- Orange window for power availability for A-train DGs removed from service during the Delta common station service transformer outage
- Orange window during midloop operations for RCS vacuum refill

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed five operability evaluations affecting risk-significant mitigating systems, listed below, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) whether the compensatory measures, if involved, were in place, would work as intended, and were appropriately controlled; and, (5) where continued operability was considered unjustified, the impact on TS Limiting Conditions for Operation (LCOs) and the risk significance in accordance with the Significance Determination Process (SDP). The inspectors verified that the operability evaluations were performed in accordance with SPP-3.1, Corrective Action Program. Documents reviewed are listed in the attachment to this report.

- PER 111520, Potentially defective external lead-wire connections in Barton pressure transmitters
- PER 109354, Safety Limit for the containment purge exhaust monitors
- PER 111178, Pressurizer concrete support cracking
- PER 116623, Axial flux difference calculation inaccuracies
- PER 116872, B-train centrifugal charging pump seal leak

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed one permanent plant modification accomplished by WO 03-019726-000, Implementation of Design Change Notice (DCN) 51370A, Replace Unit 1 Vital Inverters with New Vital Inverter Systems Including Static Switches and Regulated Bypass Transformers; Add Spare Vital Inverters. The inspectors verified that design change installation controls were adequate; that affected operational procedures and licensing documents were identified and revised accordingly; and that post-maintenance testing and equipment return to service was adequate.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed three post-maintenance test (PMT) procedures and/or test activities, as appropriate, for selected risk-significant barrier integrity and mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and, (8) equipment was returned to the status required to perform its safety function. The inspectors verified that these activities were performed in accordance with SPP-8.0, Testing Programs; SPP-6.3, Pre-Post-Maintenance Testing; and SPP-7.1, Work Control Process.

- WO 06-816442-000, Containment pressure test
- PMTI-51754-1, Steam generator post-modification test sequence
- WO 05-815593-000, 18-month sweep on 1B DG voltage regulator potentiometers.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The licensee began its U1C7 refueling outage (RFO) on September 11, 2006. Other inspection activities associated with the U1C7 outage were documented in Inspection Report 05000390, 05000391/2006004.

During this report period, the inspectors observed portions of reduced inventory and midloop operations, refueling, heatup, startup and maintenance activities to verify that the licensee maintained defense-in-depth commensurate with the outage risk plan and applicable TS. The inspectors monitored licensee controls over the outage activities listed below. Additional documents reviewed during the inspection are listed in the attachment to this report.

- Licensee implementation of clearance activities to ensure equipment was appropriately configured to support the function of the clearance
- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error
- Controls over the status and configuration of electrical systems and switchyard to ensure that TS and outage safety plan requirements were met

- Decay heat removal processes to verify proper operation and that SGs, when relied upon, were a viable means of backup cooling
- Controls to ensure that outage work was not impacting the ability to operate the spent fuel pool cooling system during and after-core offload
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss
- Reactivity controls to verify compliance with TS and that activities which could affect reactivity were reviewed for proper control within the outage risk plan
- Licensee control of containment penetrations and closure requirements in accordance with TS

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed six surveillance tests and/or reviewed test data of selected risk-significant SSCs, listed below, to assess, as appropriate, whether the SSCs met the requirements of the TS; the UFSAR; SPP-8.0, Testing Programs; SPP-8.2, Surveillance Test Program; and SPP-9.1, ASME Section XI. The inspectors also determined whether the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions.

- WO 06-817914-000, 1-SI-63-901-B, Safety Injection Pump 1B-B Quarterly Performance Test*
- WO 05-820851-000, 1-SI-61-5, 18-month Ice Condenser Lower Inlet Doors Inspection**
- WO 06-814254-000, 1-SI-30-701, Containment Isolation Valve Local Leak Rate Test, Purge Air***
- WO 05-821000-001, 1-SI-30-11-a, Containment Purge Air Cleanup System Train-A Test
- WO 05-820579-000, 1-SI-63-905, Boron Injection Check Valve Flow Test During Refueling Outages
- WO 05-820436-000, 0-SI-82-4, 18-month Loss of Offsite Power with Safety Injection Test DG 1B-B

* This procedure included inservice testing requirements.
** This procedure included ice condenser testing requirements.
*** This procedure included containment isolation valve testing requirements.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

Access Control: The inspectors evaluated licensee activities for monitoring and controlling worker access to radiologically significant areas, focusing on activities associated with the U1C7 outage, which was in progress during the onsite inspection period. The inspection included direct observation of administrative and physical controls, appraisal of the knowledge and proficiency of radiation workers and health physics technicians in implementing radiological controls, and review of the adequacy of procedural guidance and its implementation.

The inspectors reviewed licensee procedures regarding access control to radiologically significant areas. Selected procedural details for posting, surveying, and access control to airborne radioactivity, radiation area, high radiation area, locked high radiation area (LHRA), and very high radiation area (VHRA) locations were reviewed and discussed with cognizant licensee representatives. The inspectors reviewed administrative guidance documents and procedures for control of radioactive material stored in the spent fuel pool, and evaluated selected radiation work permits (RWPs) used for work in radiologically significant areas associated with the U1C7 outage. The selected RWPs were assessed for adequacy of access controls and specified electronic dosimeter alarm setpoints against expected work area dose rates and work conditions. Access control procedures for posted LHRA and VHRA locations were reviewed and discussed with selected radiation protection (RP) management, supervision, and technicians.

During facility tours, the inspectors evaluated selected radiological postings, barricades, and surveys associated with radioactive material storage areas and radiologically significant areas within the reactor containment building, reactor auxiliary building, and fuel handling building. The inspectors conducted independent dose-rate measurements at various building locations and compared those results to licensee radiation survey map data. The inspectors independently assessed implementation of LHRA controls, and evaluated the adequacy of the licensee's LHRA and VHRA key controls through procedural reviews and supervisory interviews.

During the inspection, the proficiency and knowledge of the radiation workers and RP staff in communicating and applying radiological controls for selected tasks were evaluated. The inspectors attended RWP briefings for selected work activities. Radiological worker and RP technician training/skill levels, procedural adherence, and implementation of RWP-specified access controls, including those associated with changing radiological conditions, were observed and evaluated by the inspectors during selected job site reviews and tours within the licensee's radiological control area. In addition, the inspectors interviewed selected management personnel regarding radiological controls associated with U1C7 outage activities.

RP activities were evaluated against the UFSAR Section 12, Radiation Protection; Technical Specification Sections 5.7, Procedures and Programs, and 5.11, High Radiation Area; 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. The procedures and records reviewed are listed in the attachment to this report.

Problem Identification and Resolution: PERs associated with access control to radiologically significant areas, radiation worker performance, and RP technician proficiency were reviewed and assessed. The PERs listed in the attachment to this report were reviewed and evaluated in detail during inspection of this program area. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with approved CAP procedures.

The inspectors completed 21 of the required 21 samples for IP 71121.01. All samples have now been completed for this IP.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA): Implementation of the licensee's ALARA program during U1C7 outage was observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. Reviewed activities included temporary shielding of various areas in containment, emergent work associated with the repair and test of an isolation valve (1-ISV-62-912) located in the Unit 1 Mixed Bed Demineralization - 1B pit, removal of the full-face camera in the spent fuel transfer canal, and work activities associated with the removal of the old SGs and the old storage facility. Also, incorporation of planning, established work controls, expected dose rates and dose expenditure into the ALARA pre-job briefings and RWPs for those activities were reviewed. The inspectors directly observed and evaluated the licensee's use of engineering controls, low-dose waiting areas, and on-the-job supervision.

Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting implementation of the ALARA program goals. Shutdown chemistry program implementation and the resultant effect on containment and auxiliary building dose rate trending data were reviewed and discussed with cognizant licensee representatives.

Trends in individual and collective personnel exposures at the facility were reviewed. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. The inspectors examined the dose records of all declared pregnant workers during 2005 to 2006 to evaluate total or current gestation dose. The applicable RP procedure was reviewed to assess licensee controls for declared pregnant workers. Trends in the plant's three-year rolling average collective exposure history, outage, non-outage, and total annual doses for selected years were reviewed and discussed with licensee representatives. The licensee's dose goal for the outage was 287.1 rem with a stretch target of 291 rem; however, due to an extension of the outage by 164 hours (~6.8 days) and emergent work activities, the licensee was not on target to meet their dose goal. At the time of the onsite inspection the licensee was averaging ~2.5 rem per day.

The licensee's ALARA program implementation and practices were evaluated for consistency with UFSAR Chapter 12, Sections 1-5, Radiation Protection; 10 CFR Part 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and licensee procedures. Documents reviewed during the inspection of this program area are listed in Section 2OS2 of the report attachment.

Problem Identification and Resolution: The inspectors reviewed CAP documents listed in Section 2OS2 of the report attachment that were related to the ALARA program. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Standard Programs and Processes - 3.1, Corrective Action Program, Revision 7.

The inspectors completed 25 samples for IP 71121.02 (minimum sample size is 15; however, additional line items were completed since the licensee was in the 4th quartile three-year rolling average for occupational collective dose ranking). All samples have now been completed for this IP.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Groundwater Monitoring: The inspectors discussed current and future programs for monitoring onsite groundwater with cognizant chemistry representatives, including number and placement of monitoring wells and identification of plant systems with the most potential for contaminated leakage. The inspectors reviewed and evaluated procedural guidance for identifying and assessing onsite spills and leaks of contaminated fluids. In addition, the inspectors reviewed the licensee's 10 CFR Part 50.75(g) file and compared the contents with known contaminated spill locations. The inspectors also reviewed selected parts of the 2005 Annual Effluent and Waste Disposal Report dealing with abnormal releases. Documents reviewed are listed in the attachment to this report.

The inspectors completed three of the required line-item samples described in IP 71122.01.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification & Resolution of Problems

.1 Review of Items Entered into the CAP

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing daily PER summary reports and attending daily PER review meetings.

b. Findings

No findings of significance were identified.

.2 Annual Sample: Inadequate Procedure for Containment Closure (NCV 05000390/2005004-01)

a. Inspection Scope

The inspectors reviewed the implementation of corrective actions for NCV 05000390/2005004-01, Inadequate Procedure for Containment Closure, which was documented in Level B PER 79310.

b. Assessments and Observations

No findings of significance were identified. The inspectors reviewed the licensee's root cause determination and corrective actions based on that determination. The licensee combined a Task Analysis, Why Staircase, Barrier Analysis, and Change Analysis into an Event and Causal Factor chart to determine the root cause. The root causes were determined to be: (1) the more restrictive requirement to close containment prior to core boiling in NUMARC 91-06 was not recognized; (2) calculation WBNAPS2-092 contained invalid "open containment" pressurization profiles; and (3) the NRC GL 88-17 requirement that containment closure be achieved by a valve or blind flange was not controlled by design output.

The inspectors reviewed the planned corrective actions and determined that the actions appeared adequate to correct the identified deficiencies. Eight corrective actions were implemented to address the three identified causes. The corrective actions consisted of: (1) issuing Engineering Design Calculation 51989 to revise calculation WBN-OSG4-0233, Midloop Design Information (covers multiple corrective actions); (2) issue an administrative change DCA S-38264-A to revise drawing 1-47A472-1, Mechanical Penetration Seal Details; (3) revising WBNAPS2-092, Containment Response to Loss of RHR During Mid-Loop Operation, and (4) revising TI-68.002, Containment Penetrations and Closure Control.

The inspectors reviewed the completed corrective actions and determined that the completed actions adequately addressed the identified causes with one exception. The corrective action to revise Drawing 1-47A472-1, Mechanical Penetration Seal Details (which is for auxiliary building seals) incorrectly stated test HT-EOI-24 qualified foam seals at 3 pounds per square inch gauge (psig) for 50 minutes and 2 psig for 2 hours. The corrective action justification stated that the tests for penetration X-117 did not invalidate the HT-EOI-24 tests. However, the test of X-117 demonstrated that the seals leaked at pressures as low as 0.5 psig which did invalidate the HT-EOI-24 tests for the licensee's sleeved penetrations. The test of a penetration X-117 full scale mockup, in May 2005, was in response to an NRC issue associated with penetration qualifications and test HT-EOI-24. The inspectors determined that, although the drawing change was incorrect, the operability of the auxiliary building secondary containment envelope (ABSCE) was not affected due to the extremely low differential pressures (~0.01 psig) seen by the foam seals and the ABSCE has been demonstrated operable by licensee surveillance testing. The licensee initiated PER 111908 to track and resolve this discrepancy.

.3 Annual Sample: Review of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the operator workaround (OWA) program to verify that OWAs were identified at an appropriate threshold, were entered into the CAP, and that corrective actions were appropriate and timely. Specifically, the inspectors reviewed the licensee's workaround list and repair schedules, performed CAP word searches, conducted tours and interviewed operators about required compensatory actions. Additionally, the inspectors looked for undocumented workarounds, reviewed operator deficiency lists, reviewed appropriate system health documents and reviewed PERs related to items on the workaround list.

b. Assessments and Observations

No findings of significance were identified. The inspector's review focused on two risk significant OWAs. The first was associated with the need for periodic monitoring and venting of the ERCW discharge headers so that the motor and turbine driven auxiliary feedwater (AFW) pumps would not lose suction under certain conditions. The second was associated with the need to periodically vent the residual heat removal (RHR) discharge header to prevent lifting of the RHR discharge header relief valves. Both had been classified as priority 3 OWAs (lowest priority). Per OPDP-1, Conduct of Operations, a priority 3 OWA is one that an "Operator is taking compensatory actions such as additional monitoring or leak cleanup." The inspectors questioned the licensee whether the two OWAs in question were improperly classified and should be classified as priority 2 OWAs. Per OPDP-1, a priority 2 OWA is one that an "Operator must take compensatory action during normal operations. Plant equipment could be degraded if compensatory actions not taken." The licensee initiated PER 112010 to address the inspector's question of whether the OWAs were properly classified. Long-term OWAs are a factor in determining system health in accordance with licensee procedure NEDP-12, System and Component Health, Equipment Failure Trending. The inspectors determined that incorrectly classifying the OWAs was minor because it did not affect the MR classification nor would any additional corrective actions be required.

During the development of the corrective action plan for PER 112010, licensee management agreed that, in both cases, the OWAs were improperly classified as priority 3 versus 2. The inspectors reviewed the completed PER 112010 which concluded that the ERCW venting OWA was not a workaround because the deficiency was being incorporated into the plant's design basis and the RHR venting was not a workaround since it was a singular occurrence. The inspectors questioned the licensee on the PER resolution since the original question had not been answered and the RHR venting was not a singular occurrence. The licensee determined that PER 112010 was improperly closed and initiated PER 116367 to address the deficiencies in PER 112010. The inspectors determined that closing PER 112010 without addressing the problem was minor because corrective actions to address the material issues causing the OWAs were already in place and PER 116367 was to address a process issue. The corrective action plan associated with PER 116367 was still under development at the end of the inspection period.

.4 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on human performance trends, licensee trending efforts, and repetitive equipment and corrective maintenance issues. The inspectors also considered the results of the daily inspector CAP item screening discussed in Section 4OA2.1. The inspectors' review nominally considered the six-month period of July through December 2006, although some examples expanded beyond those dates when the scope of the trend warranted.

b. Assessment and Observations

There were no findings of significance identified. The inspectors observed that the licensee performed adequate trending reviews. The licensee routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in the CAP data. The inspectors compared the licensee's process results with the results of the inspectors' daily screening and did not identify any discrepancies or potential trends in the CAP data that the licensee had failed to identify. The inspectors identified the following areas of interest during this trend review:

- As documented in Inspection Report 05000390, 05000391/2006003, paragraph 4OA2.2, the licensee had been operating with a large number of temporary alterations during the past fuel cycle. Twenty-seven temporary alterations were in effect at the beginning of the U1C7 refueling outage. A CAP review of calendar year 2006 revealed eight NRC-identified PERs involving deficiencies in the installation, administration, and implementation of the temporary alteration process. The licensee took appropriate action for each individual PER.

- A review of NRC-identified PERs for the six-month period from July to December 2006 revealed eight PERs where interim corrective actions were inadequate or not taken in time to prevent a repeat occurrence identified by the NRC. The licensee took appropriate action for each individual PER.

The inspectors discussed these observations with the licensee. The licensee initiated PERs to evaluate these observations.

4OA3 <u>Event Followup</u>

.1 <u>Notification of Unusual Event Due (NOUE) due to Reactor Coolant System (RCS) Leakage Greater than 10 gallons per minute</u>

a. <u>Description</u>

On November 21, 2006, at approximately 6:15 a.m., the shift manager declared an NOUE due to indications that RCS leakage was greater than 10 gpm. The crew had just completed a vacuum refill of the RCS, shut the pressurizer power operated relief valve, and increased RCS pressure to approximately 60 psig. After pressurizing the RCS, the crew noticed that the volume control tank level was dropping indicating leakage. The crew entered AOI-6, Small Reactor Coolant System Leak, and dispatched operators to the containment and auxiliary building to look for the leak. Pressurizer level was maintained at 60% through the event using normal charging. After adding approximately 1600 gallons of water, the volume control tank level stabilized. The licensee determined that there was not an RCS leak and terminated the NOUE at 7:37 a.m. The licensee initiated PER 115184 and performed a root cause investigation. The investigation concluded that the indications of leakage were due to filling voided portions of the RCS that were not filled during the vacuum refill.

b. <u>Inspection Scope</u>

The inspectors reviewed operator logs, plant computer data, completed procedures, and the root cause report to determine what occurred and how the operators responded. The inspectors verified that the operator response was in accordance with plant procedures, that the licensee properly implemented the Emergency Plan, and verified that notifications were made within the required time limits.

c. <u>Findings</u>

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 05000390/2005002-002-01: G45 Fuel Assembly Clad Damage

This LER is Revision 1 to LER 05000390/2005002-002-00 which was documented in Inspection Report 05000390, 05000391/2006002, Section 4OA3.2. The additional inspections of the affected fuel assembly have been delayed. Revision 1 of the LER was reviewed by the inspectors, and no findings of significance were identified and no violation of NRC requirements occurred. Additional inspections of the affected fuel assembly are planned and the results of the inspections will be provided in a supplement to the subject LER. This LER is closed.

.3 (Closed) LER 05000390/2006-004-00: Main Turbine High Vibration Trip

On March 30, 2006, the main control room staff manually tripped the reactor and turbine due to high turbine vibrations. The high turbine vibrations were due to the failure and ejection of a single blade on the last row, governor end of the C low pressure turbine. The blade ejected down into the condenser rupturing multiple condenser tubes. The inspectors' review of this event is documented in Inspection Report 05000390, 0500391/2006003, Sections 1R14 and 4OA3.1. No findings of significance or violations of NRC requirements were identified. This LER is closed.

.4 (Closed) LER 05000390/2006-007-00: High Range Radiation Monitors - Temperature Induced Current

On July 21, 2006, TS 3.3.3, condition G, was entered for both trains of containment high range radiation monitors when it was determined that upper and lower containment radiation monitors may not meet reliability expectations due to temperature-induced currents during some portions of postulated high energy line break scenarios inside containment. Condition G required immediate action to provide a report within 14 days in accordance with TS 5.9.8, Post Accident Monitoring Report. Since the temperature-induced current condition existed longer than 14 days required by TS to submit the report, this condition is reportable under 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by TS. The enforcement aspects of this LER are documented in Inspection Report 05000390, 05000391/2006002, Section 4OA2.3. This LER is closed.

4OA5 Other

.1 (Open) Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) - Unit 1

a. Inspection Scope

The inspectors verified Unit 1 implementation of the licensee's commitments documented in their September 1, 2005, response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors for Unit 1. The commitments included permanent modifications and program and procedure changes. Permanent modifications included installation of the sump screen assembly and emergency core cooling system (ECCS) cold leg injection throttle valve modification. Program and procedure changes were related to the TS surveillance for periodic screen inspection. The inspectors reviewed the sump screen assembly installation procedure and screen assembly modification 10 CFR 50.59 evaluation. The inspectors also reviewed the foreign materials exclusion controls and the completed quality assurance/quality control records for the screen assembly installation. The inspectors conducted a visual walkdown to verify the installed screen assembly configuration was consistent with drawings and verified the design criteria for screen gap.

b. Findings

No findings of significance were identified.

At the time of this inspection the Unit 1 permanent modification of the sump screen assembly, program, and procedure changes were implemented in accordance with the licensee's Generic Letter 2004-02 response. The ECCS cold leg injection throttle valve modification is pending completion and NRC review. The previously completed debris generation evaluation which could identify the need for additional actions is pending resolution of discrepancies identified by the licensee. Program and procedure changes have been completed. TI 2515/166 will remain open pending completion and NRC review of the ECCS cold leg injection throttle valve modification and debris generation evaluation.

.2 (Closed) NRC Temporary Instruction (TI) 2515/169, Mitigating Systems Performance Index (MSPI) Verification

a. Inspection Scope

During this inspection period, the inspectors completed a review of the licensee's implementation of the Mitigating Systems Performance Index (MSPI) guidance for reporting unavailability and unreliability of monitored safety systems in accordance with Temporary Instruction 2515/169.

The inspectors examined surveillances that the licensee determined would not render the train unavailable for greater than 15 minutes or during which the system could be promptly restored through operator action and, therefore, are not included in unavailability calculations.

On a sample basis, the inspectors reviewed operating logs, work history information, maintenance rule information, CAP documents, and surveillance procedures to determine the actual time periods the MSPI systems were not available due to planned and unplanned activities. The results were then compared to the baseline planned unavailability and actual planned and unplanned unavailability determined by the licensee to ensure the data's accuracy and completeness. Likewise, these documents were reviewed to ensure MSPI component unreliability data determined by the licensee properly identified and characterized all failures of monitored components. The unavailability and unreliability data were then compared with performance indicator data submitted to the NRC to ensure it accurately reflected the performance history of these systems. Documents reviewed are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

With only minor exceptions, the licensee accurately documented the baseline planned unavailability hours, the actual unavailability hours and the actual unreliability information for the MSPI systems. No significant errors in the reported data were identified, which resulted in a change to the indicated index color. No significant discrepancies were identified in the MSPI basis document which resulted in: (1) a change to the system boundary, (2) an addition of a monitored component, or (3) a change in the reported index color.

Evaluation of Inspection Requirements

In accordance with the requirements of TI 2515/169, the inspectors evaluated and answered the following questions:

1) For the sample selected, did the licensee accurately document the baseline planned unavailability hours for the MSPI systems?

   Yes. The licensee accurately documented the baseline planned unavailability hours for the MSPI systems in accordance with the prescribed method outlined in NEI 99-02, Revision 4.

2) For the sample selected, did the licensee accurately document the actual unavailability hours for the MSPI systems?

   Yes. The licensee accurately documented the actual unavailability hours for the MSPI systems in accordance with the prescribed method outlined in NEI 99-02, Revision 4.

3) For the sample selected, did the licensee accurately document the actual unreliability information for each MSPI monitored component?

   Yes. The licensee accurately documented the actual unreliability information for each MSPI monitored component in accordance with the guidance outlined in NEI 99-02, Revision 4.

4) Did the inspectors identify significant errors in the reported data, which resulted in a change to the indicated index color?

   No. The inspectors did not identify significant errors in the reported data that resulted in a change to the indicated index color.

5) Did the inspectors identify significant discrepancies in the basis document which resulted in: (1) a change to the system boundary, (2) an addition of a monitored component, or (3) a change in the reported index color?

No. The inspectors did not identify significant discrepancies in the basis document that resulted in either: (1) a change to the system boundary, (2) an addition of a monitored component, or (3) a change in the reported index color.

.3 Steam Generator Replacement Project (SGRP) Inspection Overview

This inspection report documents completion of inspections required by IP 50001, Steam Generator Replacement Inspection, some of which were completed in accordance with baseline inspection procedures. The table below identifies and correlates specific IP 50001 inspection requirements examined during this inspection period with the corresponding sections of this report. Specific documents reviewed are listed in the attachment to this report.

IP 50001 Section	Inspection Scope	Section of This Report
02.02.a	Engineering and Technical Support	4OA5.4, 5, 6 and 7
02.02.b, 02.03.c, 02.03.e.3	Design, Modification, and Analysis Associated with SG Lifting and Rigging	4OA5.10, 11 and 12
02.02.c, 02.03.e, f	Planning, Preparation and Implementation of Radiation Protection Program Controls	2OS1 and 2OS2
02.03.a	Preparation for Welding and NDE Activities	4OA5.8 and 14
02.03.b	Rigging and Heavy Lifting: Preparation and Procedures	4OA5.9 and 13
02.03.d	Rigging and Heavy Lifting: Containment Access	1R19 and 4OA5.15
02.04	Post-installation Verification and Testing	1R19 and 4OA5.14

.4 Engineering Preparation and Implementation for the SGRP

a. Inspection Scope

The inspectors reviewed engineering preparations including selected engineering packages, calculations, analyses, drawings, work packages and inspection reports, and design change notices (DCNs) for the SGRP in order to assess adequacy and completeness. The inspectors also held discussions with SGRP management to obtain a greater understanding of the entire project scope.

b. Findings

No findings of significance were identified.

.5 Project Management Organization and Staffing

a. Inspection Scope

The inspectors reviewed the SGRP organization's controls for contractor oversight and interface, plans for identifying and resolving non-conforming conditions, and plans for implementing quality assurance requirements in order to assess adequacy. The inspectors reviewed various documents, staffing reports, forecasts, and administrative procedures and conducted interviews with various personnel to evaluate the SGRP project management and organization.

b. Findings

No findings of significance were identified.

.6 SGRP Procedures and Documentation

a. Inspection Scope

The inspectors reviewed the Watts Bar Unit 1 Steam Generator Replacement Project Special Processes Manual which contained procedures for welding and NDE matrices, procurement and control of welding filler materials, welder performance qualification standards, general welding standards, NDE standards, post-weld heat treatment standards, weld documentation requirements, and welding procedure specifications to verify adequacy of SGRP procedures.

Other procedures reviewed and compared with regulatory requirements and codes that were utilized during the SGRP are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.7 Applicable Codes and Standards

a. Inspection Scope

The inspectors reviewed various sections of the Watts Bar UFSAR, Becthel's Special Process Manual, and various scoping documents to assure compliance with the appropriate code sections and editions.

b. Findings

No findings of significance were identified.

.8 Pre-Service Baseline Examination, Eddy Current Testing (ECT) of Replacement SGs

a. Inspection Scope

The inspectors verified compliance with NRC Regulatory Guide 1.83, TVA Watts Bar TS, and Sections V and XI of the 1995 ASME Code through the 1996 Addenda during a previous inspection. The inspectors reviewed the eddy current data, procedures and programs to determine if the SG tube ECT examination scope was sufficient to identify tube degradation and to confirm that the ECT scope was completed. Additionally, the inspectors reviewed the SG tube ECT examination scope to determine that it was consistent with that recommended in EPRI's Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6.

This review was performed in February 2006 and is documented in Inspection Report 05000390, 05000391/2006002, paragraph 4OA5.1, issued April 14, 2006.

b. Findings

No findings of significance were identified.

.9 Review of SGRP Lifting and Transportation Program

a. Inspection Scope

The inspectors reviewed the adequacy of the SGRP lifting programs as described in 24900-EP-004, Revision 0, Rigging and Transportation, to verify that it was prepared in accordance with regulatory requirements, appropriate industrial codes, and standards. In addition, the inspectors verified that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

The inspectors visually inspected the SGRP lifting equipment necessary to perform SG rigging and transport; reviewed the design evaluations/erections/uses and disassembly of the outside lift system; observed the removal of the shield building dome and concrete/steel containment vessel; and verified the SG compartment concrete and load drop protection.

The inspectors reviewed procedures and lifting documents that control the SGRP which are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.10 Haul Route Load Test and Evaluation

a. Inspection Scope

The inspectors reviewed the adequacy of the haul route evaluation, placement of temporary protection for plant commodities, and haul route upgrades required to prepare the haul route for load testing and transport of the steam generators to assure that they had been prepared and tested in accordance with regulatory requirements, appropriate industrial codes, and standards.

The inspectors discussed the results of the transport path load testing with SGRP engineering personnel in order to determine that, where minor discrepancies occurred, these areas had been properly corrected with appropriate material.

b. Findings

No findings of significance were identified.

.11 Observation of SG Lifting and Movement

a. Inspection Scope

The inspectors observed various portions of the old SGs being lifted from the SG cubicle through the temporary penetrations in the reactor building to the hydraulic trailer transporter. The inspectors also observed various portions of the replacement SGs being lifted from the hydraulic trailer transporter into containment. During these observations the inspectors performed visual inspections of the outside lift system and the hydraulic trailer transporter. For the task of rigging and movement of the SGs, the inspectors reviewed the work package and inspection reports for content, technical adequacy, and to verify that appropriate line items had been signed off and that required pre-lift equipment inspections had been performed and documented in the enclosures provided. This review was also to verify that industry experience was utilized and reflected in the procedures.

b. Findings

No findings of significance were identified.

.12 Review and Walkdown on Engineering Preparation

a. Inspection Scope

The inspectors reviewed the installation of temporary pipe restraints; modification of the existing restraints; removal of snubbers and beams; and pipe cuts in order to verify that the engineering preparation for the removal of the old SGs was in accordance with the engineering packages and drawings for the SGRP.

The inspectors performed a walk-through inspection of the containment building to observe the cut reactor coolant piping from the SG nozzles and to observe housekeeping conditions around the work area. The inspectors looked at corrective actions to verify that problems identified early in the outage associated with cleanliness, housekeeping, and control of materials and tools around the work area were corrected.

b. Findings

No findings of significance were identified.

.13 Interference Removal and Restoration

a. Inspection Scope

The inspectors observed the vicinity of all the SG cavities before the lifting operation began to make sure that the licensee had removed all the interferences and restraints. The inspectors reviewed procedures which controlled the removal and re-installation of interferences. Provisions for the temporary storage of removed interference items were also reviewed. In addition, the inspectors observed portions of the removal of interferences including piping, SG restraints, snubbers, and lateral supports.

After the installation of the replacement SGs, the inspectors observed various portions of the re-installation of various items including (but not limited to) piping, steam generator restraints, snubbers, lateral supports, and instrumentation tubing to ensure that they had been installed per the engineering drawings and procedures.

b. Findings

No findings of significance were identified.

.14 Special Procedures for Welding and NDE

a. Inspection Scope

Reactor Coolant System (RCS), Main Steam (MS) and Steel Containment Vessel (SCV) Fit Up and Welding

The inspectors conducted inspections of the fit up and welding activities including primary RCS, MS, and the SCV. Activities were compared to appropriate codes and standards, as listed previously in this report, and the Bechtel Special Process Manual.

The inspectors observed the machine welding of RCS hot-leg and cold-leg piping connections to all four SGs. The inspectors observed machining activities related to the weld joint preparation for MS pipe fit-up and the manual welding activities for the SCV. Inspectors also observed a sample of socket welding activities for SG level instrumentation lines. The inspectors reviewed the applicable welding procedures and associated procedure qualification records to verify that in-process welding was completed in accordance with ASME Section IX requirements.

The inspectors observed portions of the in-process welding activities for the following welds:

- Weld Number 1-068F-B003-01-c1, Reactor Coolant System cold leg pipe to SG3 nozzle, ASME Class 1
- Weld Number 1-068F-B002-01-c1, Reactor Coolant System cold leg pipe to SG2 nozzle, ASME Class 1
- Weld Number 1-068D-B001-02-c1, Reactor Coolant System hot leg pipe to SG1 nozzle, ASME Class 1
- Weld Number 1-068D-B002-02-c1, Reactor Coolant System hot leg pipe to SG4 nozzle, ASME Class 1
- Weld Number 1-068F-B001-01, Reactor Coolant System cold leg pipe to SG1 nozzle, ASME Class 1
- Weld Number 1-SCV-B023-1, 2/3 Side of Steel Containment Vessel, ASME Class 3
- Weld Number 1-SCV-B014-01, 1/4 Side of Steel Containment Vessel, ASME Class 3

Training and Qualification

The inspectors observed work, examined selected records, and reviewed procedures to evaluate the licensee's training and qualification efforts for welding and NDE personnel.

The inspectors also reviewed the programs and compared them with the regulatory requirements and codes that were utilized during the SGRP as discussed previously in this report.

NDE and Welding

The inspectors evaluated the licensee's welding and NDE activities related to the SG replacement by conducting an inspection of the records for calibration, examination results, fit-up, welding, certifications of personnel and materials, and NDE (including review of radiographs and liquid penetrant data.)

To verify that the radiographs showed the welds were free of rejectable indications, the inspectors reviewed a sample of the radiographs for intermediate (root, hot pass, 1/3 and 2/3) and completed RCS and SCV welds to verify proper penetrameter type, size, placement, and sensitivity as well as film density, identification, quality, and weld coverage.

The inspectors observed the following examinations:

Manual Ultrasonic Testing (UT):
- Weld Number RHRF-D055-11, 8" diameter valve to pipe weld, ASME Class 1

Liquid Penetrant Examination (PT):

- Weld Number 1-003B-B160-03, 3/4" Steam Generator #2 lower instrumentation tap “D”, ASME Class 2
- Weld Number 1-003B-B160-04, 3/4" Steam Generator #2 lower instrumentation tap “D”, ASME Class 2
- Weld Number 1-003B-B195-02, 3/4" Steam Generator #2 lower instrumentation tap “B”, ASME Class 2
- Weld Number 1-003B-B195-03, 3/4" Steam Generator #2 lower instrumentation tap “B”, ASME Class 2
- Weld Number 1-003B-B166-03, 3/4" Steam Generator #2 lower instrumentation tap “C”, ASME Class 2
- Weld Number 1-003B-B166-04, 3/4" Steam Generator #2 lower instrumentation tap “C”, ASME Class 2
- Weld Number 1-003B-B196-03, 3/4" Steam Generator #3 lower instrumentation tap “B”, ASME Class 2
- Weld Number 1-003B-B196-04, 3/4" Steam Generator #3 lower instrumentation tap “B”, ASME Class 2

The inspectors reviewed the following examination records:

Radiography Examination (RT):

- Weld Number 1-10, 4"-RC-14-1502, Pressurizer Spray Line, ASME Class 2
- 1-SCV-B023-01, Steel Containment Vessel (SGRP),Plate to Plate, ASME III, NE
- 1-003B-B002-17A, Feedwater - Elbow to Pipe, ASME III Class 2
- 1-003B-B002-17B, Feedwater - Elbow to Pipe, ASME III Class 2
- 1-003B-B002-08B, Feedwater - Elbow to Elbow, ASME III Class 2

Liquid Penetrant Examination:

- FSK-—156, FW-50, Residual Heat Removal (RHR) slip on flange, ASME Section III Class 2
- FSK-—200, 31" Inside diameter RCS Cold Leg for Steam Generator (SG) #3, Butt weld prep, ASME Class 1
- FSK-—198, 31" Inside diameter RCS Hot Leg for SG#1, Butt weld prep, ASME Class 1
- FSK-—017, SG#4 Shell Drain Nozzle End Preps, ASME Class 2
- Weld Number 1-068F-B001-01, SG#1 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068F-B003-01, SG#3 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B004-02, SG#3 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B001-02, SG#1 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068D-B002-02, SG#2 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068F-B002-01, SG#2 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1

- Weld Number 1-068D-B005-02, SG#4 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1
- Weld Number 1-068F-B004-01, SG#4 RCS Piping, 31" Inside Diameter (ID) Nozzle to Elbow, ASME Section III, Class 1

Magnetic Particle Examination (MT):

- FSK-—336, FW-1 for Loop 2 Upper Lateral Support, ASME Section III

Records reviewed included work packages and inspection reports, field welding check lists, filler material withdrawal authorizations, welding filler material certified material test reports, NDE reports, PT consumables certifications, quality control inspectors and NDE examiner certification and visual acuity documentation, and certification of visual acuity examiner's qualification. Records were reviewed for completeness, accuracy, and technical adequacy. The radiographs were examined for both film quality and acceptability.

b. Findings

No findings of significance were identified.

.15 Containment Restoration Activities

a. Inspection Scope

The inspectors reviewed containment restoration activities associated with two temporary construction openings in the containment shield building dome and SCV, which were approximately 20 feet by 40 feet, for the Unit 1 SGRP.

The inspectors reviewed procedures for installation of concrete reinforcing steel and Bar-Lock splices, and procedures for control of concrete placement activities. The inspectors observed installation of concrete reinforcing steel and installation of Bar-Lock splices to determine if the work was completed in accordance with requirements shown on design drawings. The inspectors reviewed results of quality control acceptance testing performed on materials (cement, fine and coarse aggregate, water, and admixtures) selected for batching the concrete, and the results of qualification testing of the Bar-Lock splices. The inspectors also reviewed the concrete mix data to ensure that selected trial mix met concrete design strength requirements, and quality control acceptance criteria, specified in the procedures for the plastic concrete, were based on the trial mix data.

The inspectors reviewed the restoration of the steel containment vessel welding procedures and weld process sheets. The inspectors observed in-process welding activities, reviewed the licensee’s program for control of weld filler materials, and observed preliminary preparations for performance of radiographs of the completed welds on the SCV. Welding acceptance criteria were specified in Bechtel General

Welding Standard GWS-1 and the American Welding Society AWS D1.1,Structural Welding Code. The following welds were reviewed for the repair to the SCV:

- Weld Number 1-SCV-B023-1, 2/3 Side of Steel Containment Vessel, ASME Class 3
- Weld Number 1-SCV-B014-01, 1/4 Side of Steel Containment Vessel, ASME Class 3

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. M. Skaggs and other members of licensee management at the conclusion of the inspection on January 10, 2007. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. One proprietary document was identified. Proprietary information was reviewed during the inspection but was not included in this report.

4OA7 Licensee-Identified Violations

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

- TS 3.6.6 requires that two containment spray trains be operable in plant modes 1, 2, 3, and 4. A plant shutdown to Mode 5 is required if one train of containment spray is inoperable for more than 72 hours. In October 2006, the licensee identified that the containment sump to B-train containment spray pump suction flow control valve 1-FCV-72-45 would not open electrically as required for B-train containment spray operability. The licensee determined that the condition had existed since March 2005. This is identified in the licensee’s corrective action program as PER 111128.

  The finding affected both the Barrier and Mitigating Systems Cornerstones and therefore required a phase 2 significance determination process (SDP). The results of the phase 2 SDP required further evaluation by a regional senior reactor analyst (SRA). The regional SRA performed an SDP Phase 3 for the finding. As the licensee only credited the use of the spray system to refill the RWST for small LOCAs, the SRA used the NRC's risk model to extract the high worth sequences for small LOCA, and for reactor coolant pump seal LOCA. These sequences were adjusted to allow refill of the RWST through the spray system, and the worth of the loss of this function was calculated. Using the maximum exposure time for SDP of one year, the finding was determined to be GREEN, and of very low safety significance.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

B. Briody, Maintenance and Modifications Manager
M. DeRoche, Site Nuclear Assurance Manager
D. Feldman, Training Manager
J. Hinman, Manager of Projects
A. Hinson, Site Engineering Manager
G. Laughlin, Plant Manager
P. Sawyer, Radiation Protection Manager
M. Skaggs, Site Vice President
J. Smith, Licensing and Industry Affairs Manager
S. Smith, Operations Superintendent
D. White, Operations Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000390/2006005-01	URI	Potential failure to properly reactivate RO/SRO licenses (Section 1R11)

Opened and Closed

None

Closed

05000390/2005-002-01	LER	Fuel Assembly G45 Clad Damage (Section 4OA3.2)
05000390/2006-004-00	LER	Main Turbine High Vibration Trip (Section 4OA3.3)
05000390/2006-007-00	LER	High Range Radiation Monitors - Temperature Induced Current (Section 4OA3.4)
05000390/2515/169	TI	Mitigating Systems Performance Index Verification (Section 4OA5.2)

Discussed

05000390/2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) - Unit 1 (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather

TI-10.17, Freeze Protection Program
SPP-10.14, Freeze Protection

Section 1R06: Flood Protection Measures

Design Criteria WB-DC-40-29, Flood Protection Provisions
Drawing 1-47W853, Flow Diagram Station Drainage System
AOI-7, Maximum Probable Flood
MI-17.018, Flood Preparation - High Pressure Fire Protection System Spool Piece
MI-17.019, Flood Preparation - Auxiliary Charging System Spool Piece
MI-17.021, Installation of Spool Pieces Between ERCW System and Component Cooling System
MI-17.022, Flood Preparation - Installation of Spool Pieces Between SFPC System and RHR System

Section 1R08: Inservice Inspection (ISI) Activities (71111.08)

Procedures, Calculations, Examination Reports
Welding Procedure Specification P8-T(RA), Revision 0 and 1
Welding Procedure Specification P1-AT-Lh (CVN 0° F), Revision 0, 2,3, and 4
Welding Procedure Specification P1-A-Lh (CVN 0° F), Revision 0, 2,3, and 4
VT - ASME III 1992, Nondestructive Examination Standard, Visual Examination, Revision 1
VT - ASME III NE, Nondestructive Examination Standard, Visual Examination, Revision 0
PT (SR)-ASME, Nondestructive Examination Standard, Liquid Penetrant Examination, Revision 0
RT-ASME III NE, Nondestructive Examination Standard, Radiographic Examination, Revision 0
RT-ASME III 1992, Nondestructive Examination Standard, Radiographic Examination, Revision 1
MT-ASME, Nondestructive Examination Standard, Magnetic Particle Examination, Revision 0
GWS-1, General Welding Standard, Revision 5
IEP-200, Qualification and Certification Requirements for TVA (TVAN) Nondestructive Examination (NDE) Personnel, Revision 8
N-UT-64, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Revision 9
1-TRI-0-10, ASME Section XI ISI/NDE Program, Revision 14
SPP-9.7, Corrosion Control Program, Revision 13
TI-68.017, Reactor Building Post Shutdown Walkdown, Revision 7
TI-100.010, System Pressure Testing, Revision 13
TI-100.009, ASME Section XI System Pressure Testing Program Basis Document, Revision 8

Work Orders (WO) and Design Change Notice (DCN)
WO 05-816062-003, Removal and Reinstallation of RCS Piping for SG3
DCN 51729, Temporary Interference removal for SG replacement

Corrective Action Documents
PER 110772, Repeat Boric Acid Leakage from Test Tee Fitting
PER 110774, Wet Boric Acid Leakage from BA Transfer Valve
PER 105098, Leak from Normal Charging Bypass Check Valve

PER 113028, Wrong welding procedure and wrong amperage downslope time used by welders (NCR-095)
PER 113705, Nonessential welding parameter violated during welding of SCV (NCR 115)
PER 112910, PT Indications identified (NCRs 084 and 091)
PER 112482, Arc Strikes on RHR flange (NCR 079)
PER 61665, Galvanic Corrosion of Emergency Raw Cooling Water bolting

Section 1R11: Licensed Operator Requalification

Procedures
TRN-11.4, Continuing Training for Licensed Personnel, Rev. 11
TRN-11.8, Operator License Examinations and Renewals. Rev. 5
TRN-11.9, Simulator Exercise Guide Development and Revision, Rev. 3
TRN-11.11, Requalification Periodic Written Examination Development and Implementation, Rev. 5
TRN-11.12, Job Performance Measures Development, Administration, and Evaluation Manual, Rev. 3
TRN-12, Simulator Regulatory Requirements, Rev. 5

Simulator Scenarios
3-OT–SRE-0022A, Simulator Examination Guide, Feedwater Malfunction following by Large Break LOCA. Rev. 1.
3-OT–SRE-0030, CCS/BREAK/ATWS/LOCA, Simulator Examination Guide, Rev. 3.

LER's
2006-003-000, Report Date, 04/03/2006, Fire Protection Header Found Isolated
2006-002-000, Report Date, 04/10/2006, Potential Loss of Cooling to the CVCS Seal Water Heat Exchanger during an Appendix R Fire
2006-001-00, Report Date, 03/14/2006, Entrained Air in Essential Raw Water (ERCW) Piping.
2005-002-00, Report Date, 01/13/2005, G45 Fuel Assembly Clad Damage (Preliminary Report.)
2005-001-0, Report Date, 05/10/2005, Two Trains of ABGTS (Auxiliary Building Gas Treatment System)
2004-002-00, Report Date, 11/18/2004, Manual Reactor Trip to Dropped Rods. 3 min 34 seconds after dropping 4 rods, the reactor was tripped

Simulator Transient Test Evaluations
2006 Transient Test 1, Manual Reactor Trip, Annual. TT1
2006 Transient Test 5, Single Loop Loss of Flow, Annual.
2006 Transient Test 8, Maximum Size LOCA with LOOP, Annual.

Malfunction Testing
Malfunction Test, IA02 - Loss of Non-Essential Control Air. Four Year Periodic. Performed on 02/02/2004
Malfunction Test, I03 - Loss of Essential Control Air, Four Year Periodic. Performed on 01/14/03
Steady State 100%/75%/50%, Simulator Regulatory Requirements, Annual. Performed on 02/22/06

Malfunction Test, CV09 - VCT Level Transmitter Lt-62-103A Failure. Four Year Periodic. Performed 02/01/04
Malfunction Test, TH04 - Pressurizer Safety Valve Failure, Four Year Periodic, Performed on 02/03/04. THIS IS 50% VS 100%.
Malfunction Test, RC-07, Pressurizer PORV Fail to Open Position. Four Year Periodic, Performed on 01/15/03

PERs (Evaluation Reports) Reviewed
13755-010-001, Resolution of Required Action to Deal with Temperature Induced Currents in Coaxial Cables
72864, Medical Restriction Notification, 11/30/2004

Program Self Assessment Reports
Self-Assessment Report SA-TRN-05-004
Self-Assessment Report SA-TRN-05-005, Assessment Date, 01/17-21/05, Report date 09/27/05 revision 2.
Self-Assessment Report SA-TRN-06-001, Assessment Date, 02/13-17/05
Self-Assessment Report SA-TRN-06-061, Assessment Date, 06/05-30/06
LT17 060613 800 - Nuclear Assurance - Watts Bar Nuclear - Audit Report No. WBA0601 - Operations Functional Area Audit.

Job Performance Measures
3-OT-JPMR108A, Return PRM –42 To Service Per AOI-4, Rev. 1
3-OT-JPMR039, Start A Thermal Barrier Booster Pump Per SOI-70.01
3-OT-JPMS090, Classify The Event Per The REP (Loss of AC Power), Rev. 6
3-OT-JPMR093, Establish RCS Bleed Paths Per FH-H.1, Rev. 6
3-OT-JPMA020B, Perform Boration Of The RCS (Locally) Per AOI-27, Rev. 3
3-OT-JPMA015A, Local Operation of Turbine Driven AFW Pump Per SOI-3.02, Rev. 4

Miscellaneous
2005 Biennial Licensed Operator Written Examinations and Results
4 Licensed Operator Reactivation Records
6 Licensed Operator Medical Files

Section 1R12: Maintenance Effectiveness

Technical Operability Evaluation, 1-98-031-9036, Temperature Control Valve, 1-TCV-67-158, Manually Positioned in the Full Open Position
WO 06-817004-001, Disassemble, Inspect, Repair and Reassemble TCV to Determine Problem with Valve Operation
PER 106173, B-train shutdown boardroom chiller tripped on 6/28/06

Section 1R15: Operability Evaluations

Calculation WBNAPS3-078, Offsite and Control Room Dose Due to a ECCS Leak Outside Containment Following a LOCA
Calculation WBNAPS3-106, Evaluation of a 3.7 gpm Leak in the Auxiliary Building

Section 1R20: Refueling/Outage Activities

General Operating Instruction (GO) -10, Reactor Coolant System Drain and Fill Operations
GO-6, Unit Shutdown from Hot Standby to Cold Shutdown
GO-5, Unit Shutdown from 30% Reactor Power to Hot Standby
TACF 1-06-0015-030, ABGTS Operation Druing Movement of Irradiated Fuel in the Auxiliary and Reactor Buildings
TI-5.009, Alternate Ventilation for Spent Fuel Pool or Fuel Transfer Canal
Calculation WCG-1-1945, Evaluation of Shield Building Dome Access Openings
Calculation WBN-OSG-40233, Midloop Design Information

Section 2OS1: Access Control to Radiologically Significant Areas

Procedures, Instructions, Guidance Documents, and Operating Manuals
Tennessee Valley Authority (TVA), TVA Nuclear (TVAN) Standard Department Procedure (SDP), Standard Programs and Processes (SPP) - 3.1, Corrective Action Program, Revision (Rev.) 11
SPP - 5.1, Radiological Controls, Rev. 5
TI-7.005, Storage of Material in the Spent Fuel Pool, Cask Pit, and New Fuel Vault, Rev. 24
TVA, TVAN, SDP, Radiation Control Departmental Procedure (RDCP) - 1, Conduct of Radiological Controls, Rev. 2
TVA, TVAN, SDP, RCDP - 3, Administration of Radiation Work Permits (RWPs), Rev. 2
TVA, Watts Bar Nuclear Plant (WBNP), Radiation Control Instruction (RCI) - 100, Control of Radiological Work, Rev. 28

Records and Data Reviewed
Radiation Work Permit (RWP) No. 06008181, U1C7 outage - keyway activities, Rev. 1
RWP No. 06009602, U1C7 outage - locked high radiation areas, Rev. 0
RWP No. 06105310, Steam generator replacement (SGR) activities: lift, transport, and store, Rev. 0
RWP No. 06105313, SGR decon LHRA activities, Rev. 0
RWP No. 06105316, SGR transport, off-load, and transfer pipe end decon spent media drums - LHRA, Rev. 1
RWP No. 06105317, SGR rigging/support, operation of cranes, movement of material in yard, Rev. 0
RWP No. 06105406, Containment SGR activities, Rev. 3
U1C7 Refueling/SGRP Outage Report, 10/06/06

Corrective Action Program (CAP) Documents
Focused Self-Assessment Report WBN-RP-05-003, Radiation Exposure Control, 08/2005
Problem Evaluation Report (PER) No. 87679, Radiological worker practices
PER No. 91647, LHRA challenges
PER No. 93317, Key control documentation
PER No. 93324, ED alarm audibility in high-noise areas
PER No. 106200, HRA posting issues
PER No. 110573, ED dose alarm
PER No. 110680, ED dose rate alarm

PER No. 110750, ED dose rate alarm
PER No. 110888, Transfer canal survey
PER No. 112297, Delay of Steam Generator #3 placement
TVA Nuclear Assurance - Radiological Protection and Control Audit Report SSA0502, 01/19/2006

Section 2OS2: ALARA Planning and Controls

Procedures, Instructions, Guidance Documents, and Operating Manuals
ALARA Pre-Planning Report (APR) 06-023, Work Associated with Reactor Pressure Vessel Head Assembly and Disassembly and Associated Refuel Floor Support Activities
APR 06-106, SGR Insulation Work Including Encapsulation
APR 06-107, SGR Scaffolding
APR 06-108, SGR RCS System Work
APR 06-109, SGR Steam Generator Secondary Side Work
APR 06-110, SGR Rigout OSG, Transport OSG, Rig in RSG including work at the OSGSF
APR 06-111, Install/Remove Pie Plate and Tripod Shielding to include Pipe End Decon
APR 06-112, SGR Radiography
APR 06-113, Install/Remove SGR QA/QC Inspections
APR 06-115, Temporary Shielding Activities
APR 06-018, Repair of 1-ISV-62-912 Isolation Valve in the Mixed Bed Demineralization - 1B Pit
RCI-128, ALARA Program Implementation, Rev. 6
TVA, TVAN, SDP, SPP - 3.1, Corrective Action Program, Rev. 11
TVA, TVAN, RCTP - 105, Personnel Inprocessing and Dosimetry Administrative Processes, Rev. 0
TVA, TVAN, SPP, SPP - 5.2, ALARA Program, Rev. 2
TVA, WBNP, Chemistry Manual, Chapter 5.09, Shutdown Primary Control Chemistry, Rev. 14

Records and Data
Active Hot Spot Database Report, Dated 10/03/06
Dose Records of all declared pregnant workers (4) during the period 05/01/2005 to present
RWP No. 06009165, U1C7 757" Aux Building, Fuel Handling, Fuel Inspection and Can Sipping, Ultrasonic Cleaning of Irradiated Fuel, Hot Particle Control Zone
RWP No. 06037132, U-1 Mixed Bed Pit to Replace Bonnet and Perform Maintenance on Valve 1-62-912 and Reach Rod and Perform Radcon Support and Surveys
RWP No. 06105310, SGR - Rig Out of OSG, Down Ending Rig Activities Transport & Store (Contaminated and Non-Contaminated Areas)
RWP No. 06105311, SGR Weld Shield Plate Covers Onto Old Steam Generator Primary Legs - LHRA (Contaminated and Non-Contaminated Areas)
RWP No. 06105314, SGR Radiography and All Associated Work in the Auxiliary Building and Outside Areas
RWP No. 06105408, U1 Containment SGRP - SGR FOSAR, Remove/Refill Water in Intermediate Leg, Remove NSG Bowl Protection, and Bowl Closeout, Primary Manway Install and Associated Work - LHRA
RWP No. 06105414, U1 Containment SGRP - SGR Rig Out OSG and Rig in RSG and Associated Work
RWP No. 06105415, U1 Containment SGRP - Rigout OSG - LHRA Activities
RWP No. 06105418, U1 Containment SGRP - SGR Radiography

RWP No. 06105504, SGR Radiography in the Unit-1 Annulus
Watts Bar Radiological Survey, Survey No. 110906-05, Dated 11/09/06, WBN319.PCX - Blank Survey - Vertical (U1 737" Mixed Bed “B” Demin Pit
WBNP - Quarterly ALARA Committee Meeting Minutes dated 07/19/05, 08/16/05, 09/27/05,11/15/05, 12/20/05, 12/22/05, 02/02/06, 02/24/06, 03/15/06, 04/19/06, 05/24/06, 06/20/06, 07/19/06, 07/27/06, 08/16/06, 08/18/06, 08/24/06,08/25/06, 09/14/06, and 09/26/06

CAP Documents
Nuclear Assurrance (NA) - TVAN-Wide - Audit Report No. SSA0502 - Radiological Protection and Control Audit, dated January 19, 2006
PER No. 91648, Radiological work planning and control for high dose outage evolutions is not always effective at implementing ALARA principles
PER No. 93788, RP personnel failed to provide an adequate dose estimate and implement adequate ALARA techniques for 729 Railroad Bay/Refuel floor material movement activities
PER No. 108983, Job of transferring the mobile demineralizer carbon bed exceeded the planned dose goal of 24 mrem by 23 mrem, historically job completed for ~20 rem

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

Procedures, Guidance Documents, and Manuals
TVA, WBNP, Chemistry Manual, Chapter 9.93, Abnormal Release Assessment, Rev. 3
WBNP, Periodic Instruction, 0-PI-CEM-2.0, Quantification of Nuclide Activity Released from Secondary Systems, Rev. 12

Records and Data
10 CFR Part 50.75(g) File
2005 WBNP Effluent and Waste Disposal Annual Report
WBN, Abnormal Release Assessment, CM, Chapter 9.93, Rev. 3, Appendix A, Abnormal Release Quantification, Auxiliary Building Exhaust, 10/22/06
WBN, Abnormal Release Assessment, CM, Chapter 9.93, Rev. 3, Appendix A, Abnormal Release Quantification, CST-A overflowing following the reactor trip on 05/30/2006, 09/04/06
WBN, Abnormal Release Assessment, CM, Chapter 9.93, Rev. 3, Appendix A, Abnormal Release Quantification, CST-A overflowing following the reactor trip on 07/31/2006, 09/04/06
WBN, Abnormal Release Assessment, CM, Chapter 9.93, Rev. 3, Appendix A, Abnormal Release Quantification, Draining of CST-A to YHP during Turbine Outage, 09/04/06

Section 4OA5: Other Activities

MSPI Verification
Watts Bar Mitigating System Performance Index (MSPI) Basis Document, Revision 0, April 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Heat Removal System, Unreliability Index, Period October 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Heat Removal System, Unavailability Index, Period July 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Cooling Water System, Unreliability Index, Period November 2006

Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Cooling Water System, Unavailability Index, Period July 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI High Pressure Injection System, Unreliability Index, Period October 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI High Pressure Injection System, Unavailability Index, Period July 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Residual Heat Removal System, Unreliability Index, Period October 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Residual Heat Removal System, Unavailability Index, Period July 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Emergency AC Power System, Unreliability Index, Period October 2006
Watts Bar Unit 1, Consolidated Date Entry 3.0 MSPI Derivation Report, MSPI Emergency AC Power System, Unavailability Index, Period July 2006
Safety System Unavailability reports for Residual Heat Removal System, High Pressure Injection System, Emergency AC Power System, and Heat Removal System for calendar years 2002, 2003, and 2004

SGRP Activities
24900-EP-004, Rigging and Transport, Rev. 0
24900-EP-006, Haul Route between RSGSA/OSGSF and Reactor Building, Rev. 0
24900-EP-008, Liebherr LR 1400 Crane , Rev. 0
Calc. No. 24900-C-001, Rev. 1, As-Rigged Weight of Steam Generators
Calc. No. 24900-C-021, Rev. 0, Haul Route Test Weight and Wheel Loads
Calc. No. 24900-C-029, Rev. 0, Evaluation of Underground Commodities Along Haul Route/Temporary Protection
Calc. No. 24900-C-006, Rev. 1, Evaluation of Safety-Related Commodities Along Haul Route for Postulated SG Drop from Transporter
DWG. 1-10W314-22, RSG/OSG Upending/Downending Foundation Location Plan, Sections and Details
DWG. 1-10W314-21, OLS Foundation Plan Sections and Details
DWG. 24900-C-002 Sht. 1, OLS Layout and SG/Heavy Lift Path, Rev. 0
Rigging International (RI) DWG. 2763-214, OLS Test Load Assembly and Rigging Details General Arrangement, Rev. 1 (Sheets 1&2)
RI DWG 2763-230, Handling SG's General Arrangement, Rev. 1
DWG 24900-C-001, RSG/OSG Haul Route Details, Rev. 0, (Sheets 3, 4 & 5)
DWG 24900-SKC-0121, Haul Route - Underground Commodities and Overhead Wires Sheets 1 & 2, Rev. B
DWG 24900-C-001, Haul Route - Plan, Rev. 1, (Sheet 1)
Letter from Lloyd's Register, 2/10/1998, Certificate Number: 9855917, Serial Number: A8800
Aboma+Kebona Certification Letter, 2/24/1999, Certificate Number: 99/029/2403, Serial Number: A8800
All Test & Inspection Letter, 2/22/2001, SAE J987 Certification pf a PTC Containerized Ringer Crane
Mammoet Crane A8800, Maintenance Log
Bechtel General Welding Standard (GWS-1), Rev. 5
Bechtel Special Processes Standard (SPS-1), Rev. 1
Bechtel CP-3 Deviation Control Construction Procedure, Rev. 2

Bechtel Nondestructive Examination Standard, RT-ASME-III 1992, Radiographic Examination, Rev. 2
Bechtel Nondestructive Examination Standard, RT-ASME-III NE, Radiographic Examination Rev. 0
Bechtel Nondestructive Examination Standard, RT-ASME-IX, Radiographic Examination, Rev. 0
Bechtel Nondestructive Examination Standard, MT-ASME, Magnetic Particle Examination, Rev. 0
Bechtel Nondestructive Examination Standard, VT - ASME III 1992, Visual Examination, Revision 1
Bechtel Nondestructive Examination Standard, VT - ASME III NE, Visual Examination, Revision 0
Bechtel Nondestructive Examination Standard, PT (SR)-ASME, Liquid Penetrant Examination, Revision 0
Bechtel Nondestructive Examination Standard, RT-ASME III 1992, Radiographic Examination, Revision 1
Bechtel Nondestructive Examination Standard, MT-ASME, Magnetic Particle Examination, Revision 0
Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, N-UT-64, Rev. 9
PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, PDI-UT-2, Rev. C
IEP-300, Qualification and Certification of Ultrasonic TVA Nuclear Personnel for Pre-Service and Inservice ASME XI Examinations, Rev. 2
IEP-200, Qualification and Certification Requirements for TVA Nuclear Personnel Nondestructive Examination (NDE) Personnel, Rev.8
Welding Procedure Specification P8-T(RA), Revision 0 and 1
Welding Procedure Specification P1-AT-Lh (CVN 0° F), Revision 0, 2,3, and 4
Welding Procedure Specification P1-A-Lh (CVN 0° F), Revision 0, 2,3, and 4
Bechtel Procedure, P1-Rebar (0.87 CE), Rev. 0
Dwg 24900-C-035 Sht 2, Shield Building Dome Form Work, Rev. 1
G-Spec G-2-WBN-01, Plain and Reinforced Concrete, Rev. 7

<u>WP & IRs</u>:
R-OSGO-127, Removal and Transport of Old Steam Generators
R-RSGI-128, Transport from RSGSA and Rig in RSG 1, 2, 3, & 4
M-LL3-148, Removal and Reinstallation of Lower Lateral Support Bumper Arrangement SG 3
M-UL2-151, SG#2 Upper Lateral Support Removal & Replacement
C-MISC2-180, SGR Structural Interferences Loop #2
M-SCV14-026, Removal and Reinstallation of the Steel Containment Vessel for SG 1&4
C-CD23-037, Removal and Reinstallation of the Shield Building Dome Sections for SG 2&3
P-RCS1-137, Cutting and Welding of RCS Piping for SG 1

<u>DCNs & Wos</u>
DCN 51681, Construct Outside Lift System (OLS) Foundation for Steam Generator Replacement
DCN 51682, Steam Generator Replacement Reactor Building Structural Mods
DCN 51729, Temporary Interference removal for SG replacement
WO 05-816062-003, Removal and Reinstallation of RCS Piping for SG3

NCRs/PERs

NCR	PER	Description
027	101402	SCV structural support steel
028	101402	SCV structural support steel
030	103239	Concrete did not meet strength requirement (PTC Crane)
046	110474	SG1 Main Steam Nozzle (OD out of spec)
048	110476	SG1 Feedwater nozzle (ID out of spec)
063	111968	Feedwater weld below min wall thickness
069	112133	Loss of FME control
--	112340	Loss of control during heavy lift
--	112420	Loss of inventory control
075	113272	SCV Liner 1-4 side (pilot holes)
076	112476	SCV Liner 2-3 side (over cut into pad)
079	112482	Arc Strikes on RHR flange
082	112484	SG 2&3 enclosure plugs
--	112589	Missing lower accelerometer missing on all 4 RSGs
083	112613	Weld prep out of spec
084	112910	PT indications (SG3 RCS)
087	112787	RCS piping weld fit-up mismatch for Steam Generator 3
091	112910	SG #1 RCS HL & CL (PT indications)
092	112976	Weld fit-up mismatch (SG1 CL)
094	113274	MS weld out of tolerance
095	113028	Wrong welding procedure and wrong amperage downslope time being used by welders
098	113272	SCV Liner 1-4 side (pilot holes)
101	113214	Hydraulic snubber tubing damaged
105	113382	Embedded ground wire cut during hydro
106	113383	Arc strike found on Loop 3 RSG
110	113589	Existing FW pipe thickness
111	113765	Preheat verification of upper lateral support on 1SG2
112	113726	Arc strikes on SCV
113	113766	Enclosure Plug Hole Inspections
114	113712	PT exam identified indication greater than 1/4" diameter
115	113705	Nonessential welding parameter violated during welding of SCV

LIST OF ACRONYMS

ABSCE	auxiliary building secondary containment envelope
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AOI	abnormal operating instruction
ASME	American Society of Mechanical Engineers
BACCP	boric acid corrosion control program
CAP	corrective action program
CFR	Code of Federal Regulations

DCN	design change notice
DG	diesel generator
ECCS	emergency core cooling system
ECT	eddy current testing
EPRI	Electrical Power Research Institute
ERCW	essential raw cooling water
FSAR	Final Safety Analysis Report
gpm	gallons per minute
HPFP	high pressure fire protection
IP	inspection procedure
ISI	inservice inspection
JPMs	job performance measures
LCOs	limiting conditions for operation
LER	licensee event report
LHRA	locked high radiation area
MS	main steam
MSPI	mitigating systems performance index
MT	magnetic particle examination
NCV	non-cited violation
NDE	nondestructive examination
NOUE	notification of unusual event
NRC	U.S. Nuclear Regulatory Commission
NUREG	(NRC) technical report designation
OS	occupational radiation safety
OWA	operator workaround
PER	problem evaluation report
PMT	post-maintenance test
PS	public radiation safety
psig	pounds per square inch gauge
RCS	reactor coolant system
RFO	refueling outage
RHR	residual heat removal
RO	reactor operator
RP	radiation protection
RPV	reactor pressure vessel
RT	radiography examination
RWP	radiation work permit
RWST	refueling water storage tank
SCV	steel containment vessel
SDBR	shutdown board room
SDP	Significance Determination Process
SG	steam generator
SGRP	steam generator replacement project
SPP	standard programs and processes
SRA	senior reactor analyst
SRO	senior reactor operator
SSCs	structures, systems, or components
TI	temporary instruction

TS	Technical Specifications
TVA	Tennessee Valley Authority
U1C7	Unit 1 Cycle 7
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
VHP	vessel head penetration
VHRA	very high radiation area
VT	visual examinaiton
WO	work order
°F	degrees Fahrenheit